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THE SHIPPINGPORT PRESSURIZED WATER REACTOR

THE SHIPPINGPORT PRESSURIZED WATER REACTOR

Written by Personnel of the

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WESTINGHOUSE ELECTRIC CORPORATION
Bettis Plant

DUQUESNE LIGHT COMPANY

PREPARED FOR THE UNITED STATES
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FOREWORD

On December 18, 1957, the Shippingport Atomic Power Station began supplying electricity to the Duquesne Light Company, serving the Greater Pittsburgh, Pennsylvania, area. That date marked the successful culmination of more than four years of planning, development, and construction to build the first large-scale central station nuclear power plant in the United States and the first plant of such size in the world operated solely to produce electric power.

The decision to construct a large-scale pressurized water reactor plant came in 1953. For some years there had been a growing realization that planning for future nuclear power plants of two or three hundred megawatts electrical output could be made with greater confidence if operating experience from a reactor plant of intermediate power were available. At that time, the one large United States reactor project from which such experience might have been obtained had just been cancelled. This project (known as the CVR) was for the development of a large military pressurized water reactor for a naval ship. Consensus was reached among officials of the Atomic Energy Commission and members of the Joint Committee on Atomic Energy of the Congress that construction of a pressurized water central station plant, whose design would be extrapolated from the preliminary work carried out under the CVR project, would speed nuclear power development both in the United States and abroad.

Formally authorized in July, 1953, the civilian undertaking, often called the Pressurized Water Reactor project, got under way in its planning stages within weeks. Responsibility for the project was assigned by the Atomic Energy Commission to Rear Admiral H. G. Rickover, U.S.N., Chief of the Naval Reactors Branch, Division of Reactor Development. In July the Westinghouse Electric Corporation was chosen to design, develop, and build the reactor plant under contract to the Commission. In succeeding months, other contractors were selected and groundwork laid for fabrication of components and for starting construction.

In March, 1954, from nine proposals submitted to the Commission to build and own the electrical generating facilities, that of the Duquesne Light Company, Pittsburgh, was selected as most favorable to the Government. Duquesne furnished the plant site at Shippingport, on the Ohio River about 25 miles west of Pittsburgh, and the turbine-generator plant that it operates at no cost to the Government. The company also assumed $5 million of the cost of the reactor portion of the plant. Al-

though Duquesne also operates and maintains the reactor plant, title to this nuclear facility remains with the Government.

By subcontract to Westinghouse, Stone and Webster Engineering Corporation provided the architect-engineer services for the nuclear portion of the plant. Dravo Corporation, of Pittsburgh, also under subcontract to Westinghouse, installed the nuclear plant. Burns and Roe, Inc. acted as agent-constructor for the Duquesne Light Company in building the turbine-generator portion of the plant.

The Naval Reactors Branch was charged with the responsibility of giving technical approval to all nuclear plant parameters, performance requirements, and details of design and development upon recommendation of the contractor, Westinghouse, working through the Commission's Bettis Plant in the Greater Pittsburgh Area. The project officer was Captain J. H. Barker, Jr., of the Naval Reactors Branch. Throughout the design, construction, and successful operation of the Shippingport plant, Admiral Rickover exercised close control and continuously made thorough inspections. A PWR Project Officer or Project Manager was named in each participating organization and committees were named to coordinate and operate the project.

Official groundbreaking took place September 6, 1954, when President Eisenhower, in Denver, closed a circuit starting an unmanned bulldozer at Shippingport—the site that had been chosen for the atomic power plant. Actual construction started in May, 1955.

After an extensive program of testing and checkout to meet stringent safety and health requirements, the Shippingport Reactor first went critical on December 2, 1957. Sixteen days later, as earlier noted, power was fed into the Duquesne system. On December 23, the Shippingport plant achieved its full power of 60,000 kilowatts net, with three loops operating.

In the following months, a series of power operation tests were conducted to determine the operating characteristics and dependability of the Shippingport station.

On May 26, 1958, President Eisenhower, speaking from the White House, dedicated the station. At the conclusion of his brief dedicatory remarks, he used the same neutron wand and neutron counter which he had employed in the ground-breaking ceremonies in 1954 to start power generation at the plant.

Particularly important in a continuing program to disseminate widely information on the Shippingport plant were: (1) publication up to this time of six volumes touching on various phases of PWR and other Naval Reactors Branch technology and plans for publication of nine more, and (2) four seminars for representatives of industry that were held in 1954 and 1955. In addition to these formal information meetings, more than

10,000 visitors have been conducted on tours of the Shippingport site and facilities since construction began.

From inception, the main purpose of the Shippingport plant has been to advance the technology of pressurized water reactors rather than to generate electricity at costs competitive with ordinary fuels. An important milestone in the peaceful use of the atom, the plant was a cooperative undertaking of the United States Atomic Energy Commission, the Westinghouse Electric Corporation, and the Duquesne Light Company.

In the pages that follow, the technology of the Shippingport plant is reported by those best qualified to do so—the scientists and engineers who designed and constructed the plant, tested it, and placed it in operation.

Contained in this book are some of the technical history and important details of the station's design, development, and construction. Of special interest to the technical reader, Chapter 1 on design philosophy summarizes the basic reasoning that went into the selection of the plant design parameters. The Shippingport reactor has a seed and blanket core with all fuel (both the enriched uranium "seed" and the natural uranium dioxide "blanket") clad in Zircaloy. The present core (called Core I) is designed for a thermal output of 225 megawatts. Net electrical power of the reactor is now 60 megawatts. However, because of the developmental nature of the plant, the turbine-generator was designed for an electrical output of 100 megawatts. Thus the power output of future cores can be increased to match generator capacity. Since the Shippingport plant is large and capable of utilizing cores of higher thermal capacity and more advanced design, its operation should add significantly to pressurized water technology during coming years. Indeed, it has already done so in the months of operation completed to date.

Chapters are also included on reactor design, fuel element development, and core construction. The many auxiliary systems required in the reactor plant are discussed in considerable detail. The water chemistry of the reactor system is discussed specifically as it applies to the Shippingport plant, but much general pressurized water reactor chemistry is presented as well. In the chapter on physics the seed and blanket core concept and its advantages are described at considerable length. Another chapter discusses quite extensively the instrumentation systems that control and protect the plant.

In addition, other chapters take up the testing program, the radioactive waste disposal system, electrical and mechanical components used in the reactor plant (most of which were either of specialized design or were larger than any corresponding equipment known to have been built previously), shielding of the reactor plant and waste disposal facilities, hazards evaluation, and a description of the turbine-generator plant.

Information also is given on the choice of the Shippingport site, design of the nuclear portion of the plant, and construction details. Of great importance are chapters describing: (1) the organization and training of personnel for operating the station and the preparation of the station for operation, and (2) the methods developed for controlling a project of such magnitude, and for procuring, under a stringent schedule, components and equipment to very rigid specifications.

The Shippingport Atomic Power Station is the result of the unstinting efforts of many people. Officials of the Atomic Energy Commission, members of the Congress of the United States, and officers of the Westinghouse Electric Corporation and the Duquesne Light Company have strongly supported and sustained the project. In particular, tribute is due Admiral H. G. Rickover, Chief of the Naval Reactors Branch of the Commission's Division of Reactor Development, whose foresight, determination, and able administration saw the project through difficulties that at times appeared insuperable, and to the engineers, scientists, and others of the Naval Reactors Branch and the Duquesne Light and Westinghouse companies whose hard work brought the Shippingport plant into being. The key staff members of these organizations who, in the midst of demanding assignments, prepared this record, have the appreciation of the Commission.

Washington, D.C.
July 1958

A. TAMMARO,
ASSISTANT GENERAL MANAGER FOR RESEARCH AND INDUSTRIAL DEVELOPMENT, UNITED STATES ATOMIC ENERGY COMMISSION

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CHAPTER 1

DESIGN PHILOSOPHY

CHAPTER 1

DESIGN PHILOSOPHY*

This chapter presents some important concepts that guided the design of the Shippingport plant. It opens with a brief description of the plant to orient the reader. The brief plant specifications established by the Atomic Energy Commission to guide the design are then presented, with discussion of some of the reasons for choosing certain values. Over-all reactor plant design philosophy and reactor design philosophy are given next, followed by the considerations governing design of reactor plant facilities.

1–1. Purposes of PWR

In addition to laying the groundwork for the design, construction, and operation of future nuclear power plants, the Shippingport plant serves a number of other purposes. As the first U. S. nuclear plant in public utility service, it constitutes an exploration of one method of building and operating a large reactor plant with the safeguards necessary near a large city. It is a facility in which a variety of designs of light water cooled and moderated reactor cores can be operated under suitable conditions. Finally, the plant is a power producer, designed to sustain its full load with high availability at all times.

1–2. Brief Plant Description

The basic schematic diagram of the PWR reactor and steam plant is shown in Fig. 1–1. The principal elements of the reactor plant are the reactor vessel, containing the nuclear core, and four main coolant loops which circulate the reactor coolant water between the core and the steam generators. With three loops operating, the reactor produces approximately 225 Mw of heat, which is converted to 60 Mw net electrical output.

The core, or heat producing unit of the reactor, is made up of two types of fuel elements assembled into a right circular cylinder. The active portion of the core, that part containing fissionable material, is about 6 ft high and about 6 ft 7 in. in diameter. The core is of the "seed and blanket" design. Highly enriched uranium, which forms the "seed," is in 1914

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*By P. G. DeHuff and W. R. Ellis, Westinghouse Bettis Plant, and I. H. Mandil and M. Shaw, U. S. Atomic Energy Commission.

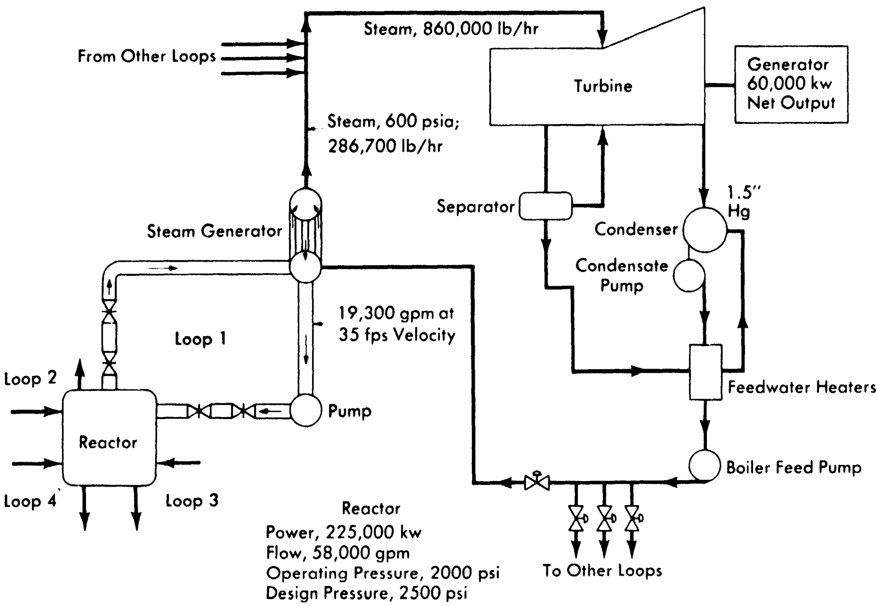

FIG. 1-1. Basic schematic diagram of the Shippingport plant.

zirconium clad plates, the active portions of which are 2.05 in. wide by 70.75 in. long. The "blanket" contains natural uranium, in the form of uranium oxide, in about 95,000 fuel elements; each element is a zirconium tube, 0.411 in. in diameter and 10.25 in. long, filled with natural uranium oxide pellets. The core and its 32 control rods are housed in a reactor vessel which measures 109 in. in internal diameter and about 32 ft in inside height.

Each of the four main coolant loops contains a single stage centrifugal canned motor pump, a steam generator, four isolation valves, a check valve, and interconnecting 18-in. piping.

High purity water, under a pressure of 2000 psi, serves as both coolant and moderator. At full power, with three loops operating, 58,000 gpm flow through the core, entering at 508°F and leaving at about 538°F. Velocity in the 18-in. pipes is approximately 35 ft/sec, and in various parts of the core, between 4.5 and 20 ft/sec. The total pressure drop around a main coolant loop is approximately 105 psi, divided about equally between the core, the steam generator, and the piping.

Heat absorbed by the reactor coolant in passing through the core is given up to the steam plant water on the secondary side of the steam generator heat exchanger to produce steam at a pressure of 600 psia at the generator outlet. This steam is delivered to an 1800 rpm turbine which, along with its connected generator, has a maximum capability of 100 Mw

gross electrical output. The steam plant equipment differs from that of a conventional plant primarily in that it is designed to handle dry and saturated steam at relatively low pressures. As a result, the steam delivered to the turbine must be extracted when its moisture reaches about 13% and run through a moisture separator for drying before it can continue its expansion cycle through the turbine. The steam cycle uses three stages of feed water heating. The condensate pumps are rated at 2000 gpm at a 300-ft head; the two boiler feed pumps are each rated at 700,000 lb/hr and 825 psi.

1–3. General Reactor Plant Specifications

When the PWR Project began in July 1953, the Atomic Energy Commission established these general specifications for design of the plant:

(1) Generation of at least 60,000 kw of useful electric energy.
(2) Use of a light water cooled and moderated, slightly enriched uranium type reactor.
(3) 600 psi saturated steam.
(4) Fuel element life as long as possible between chemical reprocessings (initial goal in excess of 3000 Mwd per ton).
(5) Refueling with minimum shutdown period.
(6) Simplified reactor control system.
(7) Central station type turbine and electrical generating equipment.
(8) Conventional central station steam, electric, and other auxiliary systems.
(9) Commercial standards of equipment.
(10) Use of concrete for shielding.
(11) Minimum construction cost of the plant.
(12) Minimum operating cost of the plant consistent with the above requirements.

Several of these general specifications, particularly items 5 and 7 through 12, were self-explanatory. Some of the thinking behind the selection of the remaining requirements was as follows:

1–3.1 Plant rating. A 60-Mw plant seemed to be a reasonable extension of the technology developed in the construction of cores and reactor plant equipment for previous pressurized water reactors. It appeared that higher capacity — for example, a jump to 150 or 200 Mw — would yield little additional information on the key design problems but would greatly increase total project costs. Moreover, it was recognized that after the plant had been built and tested, advances in reactor design technology might well demonstrate that power output could be increased with a new core design. (Later, anticipating this, the Duquesne Light Company, builder of the steam electric plant, installed a 100-Mw turbine-generator.)

1-3.2 Coolant and fuel. The Atomic Energy Commission selected the pressurized light water concept for the Shippingport plant because, at the time, it was the only type with proven feasibility for power production. Use of heavy water as moderator and coolant did not appear to offer any over-all advantage. Also, use of heavy water would have increased the cost and complicated the design of the reactor plant, since great care must be taken to prevent its loss. In 1953 a slightly enriched uranium core was tentatively chosen in an attempt to reduce power cost because a highly enriched core appeared unlikely to produce power at prices that would compete with coal. At that time the seed and natural uranium blanket core had not been conceived. This advanced physics concept was evolved in 1954 and adopted in December of that year. It was aimed at obtaining the maximum power from natural uranium, with minimum investment and expenditure of U^{235} and with a reduction in the mechanical reactivity control requirements of the core.

1-3.3 Steam pressure. Work on another project had established the feasibility of obtaining 600 psi steam at full power from a large pressurized water plant. Thus, it was thought that the Shippingport project could contribute most valuably to pressurized water reactor technology by employing the highest reasonable steam pressure, then estimated to be 600 psi. It was considered probable that future PWR plants would tend toward higher steam pressure to increase steam plant efficiency and reduce operating costs. It was decided not to try to optimize the steam pressure for PWR on the basis of estimated equipment and core costs since these could be expected to change drastically in the future.

Saturated steam was selected, since to superheat 600 psi steam with the core maximum surface temperature available was not considered feasible. This maximum temperature was set at 636°F on the basis of the then current Zircaloy corrosion data and, as explained later in this chapter, a further design requirement that coolant boiling in the core during normal steady-state conditions must be prevented. An oil-fired superheater was not adopted; although its use would have reduced the initial power cost, it would not have contributed to reactor plant technology.

1-3.4 Fuel-element life. A major factor in the cost of power from a pressurized water reactor is the expense of fabricating the core. No simple, inexpensive fuel element had yet been built for this type reactor. Therefore, it appeared that one obvious way to reduce power cost was to increase core life so that fabrication costs, which are relatively constant for a specific core design, could be spread over more kilowatt-hours. Consequently, an initial PWR design goal was a natural uranium fuel element with an average life of 3000 Mwd/t. Furthermore, long core life implied

generation and burning in place of plutonium. This was also considered a great advantage, especially since the Shippingport reactor was not intended to be a plutonium producer.

1–3.5 Reactivity control. Simplified reactivity control was required to reduce the electrical and mechanical complications of previous reactivity control schemes. It was thought that it might be possible to develop a means of chemical control that would be simple, relatively inexpensive, and reliable. Studies of chemical control schemes using poisons dissolved in the reactor coolant revealed that such reactivity control had complications that might make fail-safe characteristics difficult to achieve, and that its use in a seed and blanket core was undesirable. Therefore, study of chemical reactivity control was abandoned (except for chemical shutdown as described in Chapter 8). The PWR control scheme adopted was basically a reliable one, simple in concept, and actually produced very stable reactor plant operation.

1–4. Reactor Plant Design Philosophy

1–4.1 Over-all plant design requirements. A number of design requirements were established early in the project, with emphasis placed on plant safety and reliability. The more important requirements were:

(1) Safety must be an overriding feature.
(2) The reactor coolant must be retained in a sealed system.
(3) The materials in contact with reactor coolant must be corrosion resistant.
(4) The reactor plant must be able to continue operation with failed fuel elements in the core.
(5) A rupture of the reactor coolant pressure boundary outside the reactor vessel must be capable of isolation.
(6) All equipment handling reactor coolant must be designed and built for maximum reliability.
(7) All such equipment must be manufactured under clean conditions.
(8) High temperature, high pressure coolant must be retained inside the plant container in case of a casualty.
(9) Reactor decay heat must be dissipated without the use of an external source of power.
(10) Shielding must be provided so that repairs can be made on one reactor coolant loop while the remaining three are in operation.

Economy was an ever-present consideration. A more detailed discussion of each requirement follows.

Safety. In the design, construction, and operation of the Shippingport plant, every necessary precaution has been taken to guard against hazards

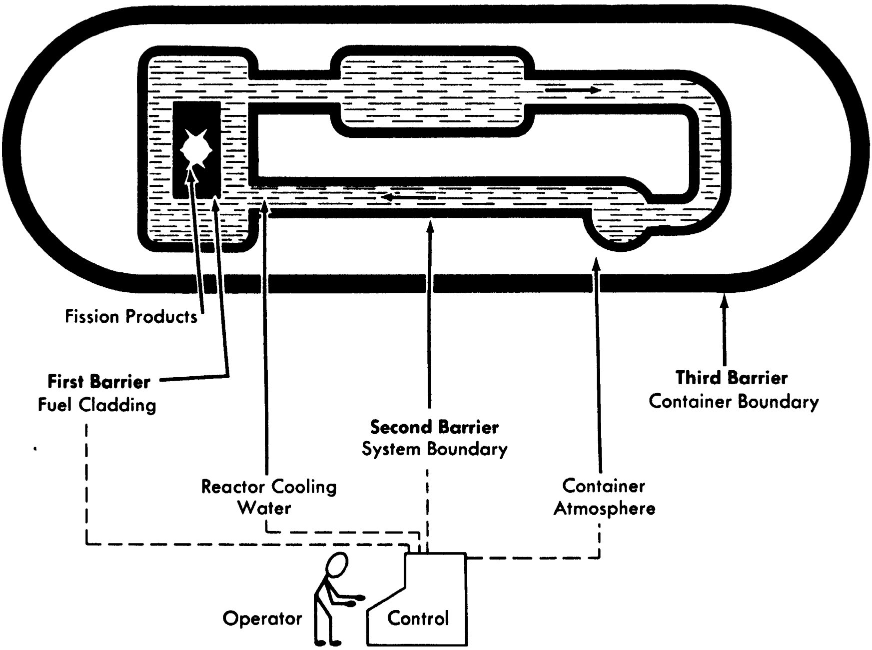

FIG. 1–2. The three independent barriers of PWR.

to the operating personnel or to the surrounding area. The principal safety problem is to prevent, in the event of any conceivable accident, the release of harmful quantities of fission products from the reactor core. Thus, PWR fission products are enclosed within three independent concentric barriers; equipment is installed to protect the reactor from damage and to prevent the release of fission products; and operators are properly trained for both normal and emergency conditions.

Of the three barriers mentioned above, the first two are continuously monitored to detect failure. The first barrier is the fuel element cladding, made of corrosion-resistant Zircaloy. The fuel elements are designed, fabricated, and inspected to ensure high integrity. The second barrier is the all-welded corrosion-resistant steel reactor coolant pressure boundary. If fission products leak into the coolant through flaws that may develop in the fuel element cladding, the coolant pressure boundary will contain them. There are provisions in the core to locate and remove faulty fuel elements. The third barrier is the reactor plant container, a series of four large interconnected, vapor-tight steel pressure vessels housing all parts of the plant that contain coolant at high temperature and pressure. This barrier system is depicted in Fig. 1–2.

The systems and equipment installed in the plant to protect the reactor

from damage and prevent release of fission products include the reactor protection system (Chapter 9), the safety injection system (Chapter 8), and the operational radiation monitoring system (Chapter 9). In fact, most systems described in Chapters 8 and 9 significantly contribute to plant safety.

Sealed system. Although the radioactivity of pure water decays rapidly, dissolved impurities, corrosion products, or fission products that have longer periods of radioactivity may be in the coolant water. Therefore, it is important to limit leakage from the reactor coolant system — for example: (1) Leakage to the plant container must be limited for the safety of the operators and the surrounding area. Such leakage would ultimately contaminate external surfaces and increase maintenance costs. (2) Leakage through the steam generator heat exchangers would increase operating and maintenance costs because small amounts of radioactivity would be transferred to the steam plant. (3) Leakage to the coolant discharge and vent system would increase water consumption, depletion of chemicals and ion exchange resin, and operating costs in the radioactive waste disposal system. Such leaks, once established, could increase with time.

Therefore, it was decided early in the project that the portions of the reactor plant containing coolant at operating temperature and pressure would be, or at least made capable of being, hermetically sealed. With a sealed system, the plant could more feasibly continue to operate even though the coolant might contain fission products from a number of minor fuel element leaks. This decision for a sealed system dictated use of canned-motor main coolant pumps and control rod drive mechanisms; seal welds on all flanged joints, including that between the reactor vessel and the reactor vessel head; seal welds on steam generator tube-tube sheet joints; capped manual valves and sealed remotely-operated valves; and sealed instruments. It also resulted in the use of double stop valves in drain lines from portions of the plant containing hot pressurized coolant. A general design rule was applied throughout the reactor coolant systems (those containing reactor coolant at operating temperature and pressure). Nowhere in these systems would pressure integrity depend solely on a packed or gasketed connection. It was considered satisfactory, however, to have a packed joint and a gasketed connection in series if the gasketed connection could be seal welded if desired.

Corrosion-resistant materials. Corrosion, a problem in any power producing facility, is a major one in a nuclear reactor, since its products may become activated and thus sources of radiation. Corrosion products may also be carried in the coolant stream and deposited on the heat transfer surfaces of the fuel elements, thus reducing heat transfer from the fuel to the coolant. This, along with the possibility that the particles may restrict coolant flow in the channels provided, could result in severe local overheating and possibly fuel element burnout.

For this reason corrosion-resistant materials are used throughout the reactor plant where they come in contact with the coolant water. Early in plant design, consideration was given to carbon steel in lieu of stainless steel for components such as the reactor vessel, main coolant piping, steam generators, and pumps. However, at the time these components were ordered, it was impossible to prove that carbon steel would be satisfactory when used in contact with high temperature reactor coolant. Rather than jeopardize plant performance, the decision was made to use entirely corrosion-resistant reactor coolant systems. Carbon steel has been used, where feasible, in handling coolant after it has been discharged from the reactor plant. Materials with adequate corrosion resistance at low temperatures handle reactor plant make-up water. For example, bronze was considered satisfactory for valves handling low temperature water in the coolant charging system.

Fission products assumed to be in reactor coolant. To improve plant load factor, the plant was designed so that it can continue to operate even if some fuel elements in the reactor develop flaws and discharge fission products into the coolant. It was assumed that about 1% (1000) of the uranium dioxide blanket fuel elements might develop defects during the life of the core. It was also assumed that of the 1000 defected elements, as many as 30 might develop major defects. These are pessimistic assumptions, since the plant contains facilities for locating and removing defective fuel elements. These assumptions were based partly on the belief that experience would probably show it to be too costly to manufacture cores so perfect that no fuel elements fail. They were also based on the fact that this plant is experimental, and may be used to test many types of experimental fuel elements. Some of these may develop small defects, or fuel elements containing deliberate defects may be tested. This decision caused some increase in plant cost; however, this additional cost should pay a dividend in reducing the cost of future cores. The principal effect on plant design was: (1) the requirement that all vents and drains from the reactor plant be led into a closed system and (2) the addition of some equipment to the radioactive waste disposal system.

Remote isolation of reactor piping connections. One of the most serious accidents to a reactor plant is accidental loss of reactor coolant, such as might result from a leak. If as a result of such an accident the core became uncovered, it might be heated to the melting point by decay heat. The result might be release of fission products to the plant container. To help prevent this "loss-of-coolant" accident, all pipe lines leading to the reactor vessel were equipped with either automatically or remotely operated isolation valves. The valves were located as close to the reactor vessel as practical so that any coolant leak that might develop in the plant equipment or piping could be isolated from the re-

actor. All main loops and the pressurizer can be so isolated by remotely-operated gate valves. Because of the tendency of pressure relief valves to remain open after excess pressure has been released, a pair is installed in parallel and an isolation valve is installed downstream of each valve. These particular isolation valves are so arranged that one is always open. Small instrument and pipe lines are protected by excess flow check valves that permit passage of normal flow, but close on excess flow that might result from a break in the pipe.

Close control of design, manufacture, and test. The Shippingport plant contains much specially designed, or newly developed, equipment. Experience in building similar "developmental" plants has shown that, for reasonable plant reliability, great pains must be taken to insure that equipment is properly designed, carefully manufactured, and, if possible, thoroughly tested before being installed. These precautions also add to plant safety.

In designing and building PWR, design and manufacture of vital equipment was closely controlled. All electrical and mechanical equipment affecting plant safety or reliability was built to rigid specifications. First, there was a detailed or performance specification outlining equipment requirements and acceptance tests. Second, there were detailed material and process specifications covering not only chemical composition, strength, and quality of the material used, but also certain design, manufacture, and inspection procedures to which adherence was required.

Since use of adequate specifications does not insure high quality components, further steps were taken. Detailed design drawings and specifications for vital and development equipment were reviewed by two independent groups of design engineers, as well as by materials and process engineers where appropriate, before being released for manufacture. In addition, frequent equipment inspections were made during manufacture. Finally, where possible, a thorough operational test of the equipment was made before it was installed.

The plant was designed to comply with the applicable ASME and State of Pennsylvania codes. Due to the special nature of nuclear power plants, special code rulings on certain features of the plant had to be obtained.

Equipment manufacture and plant assembly under clean conditions. All components handling reactor coolant before entry into or during circulation in the reactor plant were manufactured and, with associated piping, assembled under exceptionally clean conditions. Cleanliness was controlled by detailed process specifications that became a part of each component equipment specification. There are several reasons for the necessity of a clean plant. Many components in the reactor coolant system have water lubricated bearings or water lubricated sliding surfaces. For example, the bearings on the canned-motor pumps consist of Graphitar running against

a hardened corrosion-resistant steel; control rod drive mechanisms contain water lubricated ball bearings; and all valves, and some other equipment, contain many sliding surfaces. If the reactor coolant contains significant quantities of foreign matter, such as metal chips or dirt, these bearings and sliding surfaces might be damaged and make the equipment inoperative. Special precautions were also taken to eliminate oil or similar materials that might coat and reduce the performance of heat-transfer surfaces. In addition, it was considered undesirable to have in the coolant any chemical element that could become highly radioactive upon passing through the core. Such material could later be deposited in the plant and result in areas of high activity, thus hampering maintenance operations.

As a further precaution, a special filter was built and installed in the reactor vessel to collect all foreign particles from the reactor coolant systems during initial plant testing prior to core installation. These cleanliness requirements added to the cost of the plant but were believed essential if expensive repairs and plant outages were to be prevented.

Containment of reactor coolant. The reactor coolant can contain varying amounts of radioactive materials, depending primarily on whether fuel elements have failed and are releasing fission products. Studies show that release of all coolant to the atmosphere outside the reactor plant container — even with equilibrium fission products from 1000 failed fuel elements contained therein — would not present a serious hazard to personnel. However, it might slightly contaminate some of the surrounding area. Therefore, so that even this will not happen, the decision was made that no pipe or piece of equipment containing high-temperature (above 200°F), high-pressure coolant could be located outside the plant container.

Decay heat dissipation without power. The reactor core continues to generate heat many days after shutdown from power operation. The amount of such heat requires that it must be dissipated to prevent the core from melting and releasing fission products. One of the safety design criteria for the plant was that decay heat dissipation would not depend on the availability of a power source other than the reactor itself. It was further specified that decay heat dissipation must be automatic and that the plant must be capable of protecting itself without operator action for two hours after loss of all power. This was accomplished by locating the reactor vessel and the steam generators so that natural circulation of the reactor coolant would take place, delivering the heat from the reactor to the steam generators. The steam generated is then released by a relief valve on the main steam header, preventing the plant temperature from exceeding a preset value.

Repairability. Much PWR equipment is developmental. Even though steps were taken to insure its reliability, some failures of major equipment might be expected. To minimize the effect of these failures on plant

operation and availability, a spare main coolant loop was installed. So that equipment in any loop could be repaired while the plant was running, each loop was shielded separately. This feature was also carried to auxiliary equipment and instrumentation insofar as practical. Where possible, reactor plant auxiliary equipment such as the coolant charging, valve operating, component cooling water, and coolant sampling systems were located in readily accessible areas outside the plant container. This helped reduce investment and maintenance costs by permitting more use of local manual valves, local control equipment, and local instrumentation, instead of remotely operated devices.

1-4.2 Development of major plant design features. A discussion of criteria used in selecting plant features follows. Information is included on the selection of the number of main coolant loops, the average reactor coolant pressure and temperature, the amount of pumping power, and the type of reactor control.

Selection of four loops. Modern power plant design practice utilizes one boiler connected to the turbine. Thus, a major equipment failure in either unit puts the entire system out of operation. Reliability of conventional equipment, ease of repair of such equipment, and system electrical interconnection have so progressed that such a system has proven economical. It was recognized that ultimately nuclear power plant construction costs will be minimized by use of as few separate components and main coolant loops as possible. However, in present nuclear power plants, because of the longer time required for major repairs, and the fact that many items of plant equipment are presently developmental, reliability would be greatly reduced by use of only one steam generator per reactor. In addition, the size of the pumps, valves, and steam generators that would have been required in the Shippingport plant if only one loop had been used would have called for an excessive extrapolation of design and fabrication knowledge developed up to that time.

To select the number of loops, a study was made of initial cost, size of components, and plant output that would be lost with one loop out of service. The study revealed that because of the higher development costs expected with the use of the larger components required for a smaller number of loops, the total plant costs would not vary greatly with the number of loops. The effect of incapacitation of a main coolant pump was also a major consideration. With this in mind, the desirability of having two pumps per loop, using one as an installed spare, was evaluated, but the additional cost could not be justified.

Consideration was given to obtaining an installed spare pump in the form of a spare loop. This was studied for a total of three, four, and five loops. With four loops the pumps required were about four times the

capacity of the largest ones previously built. This appeared to be reasonable extrapolation. The pipe size required for four loops approached the largest that industry was equipped to build in the wall thickness required. In addition, with four loops, the loss of a component in one would result in a plant flow capacity reduction of only 15 to 20%. For these reasons, the four-loop design was adopted, one loop being considered a spare. Furthermore, it was decided that it would be preferable to design the loop layout and shielding so that major repairs, such as a pump replacement, could be accomplished in one loop with the others operating.

Selection of reactor coolant pressure. The nominal reactor coolant system operating pressure is 2000 psia. This pressure was selected as a compromise of many factors. To minimize the cost of the reactor plant it is desirable to keep the reactor coolant pressure low. However, the lower the coolant pressure, the lower must be the coolant temperature and thus the core fuel element surface temperature, if boiling in the core is to be prevented. Lower fuel element temperatures mean lower turbine plant steam pressure, which in turn means lower steam plant efficiencies and increased steam plant costs. Therefore, it was decided that rather than attempt studies to optimize the steam pressure for the PWR plant, the maximum feasible fuel element temperature and steam pressure would be used. It appeared that this would contribute most to the advancement of reactor technology, since future pressurized water plants would probably tend toward higher temperatures.

On the basis of this philosophy and all information available on the behavior of Zircaloy material in water at high temperature, a maximum fuel element surface temperature of 636°F was selected. This temperature allowed a safety factor for transients, instrument inaccuracies, and the uncertainties related to calculating fuel element hot spot temperatures. Based on this temperature and the requirement that there be no nucleate boiling in the core, the reactor coolant pressure was chosen as 2000 psia (i. e., saturation pressure for 636°F).

The design pressure was set at 2500 psia to include allowances of 200 psi for variation in operating pressure during power transients and 300 psi for use in setting the pressure of the relief valves. The allowance of 300 psi for relief valve operation permitted cascading several valves in series to give more reliable operation, as explained in more detail in Chapter 8. The 2500 psi design pressure had the added advantage that previous pressurized water reactor plants had been designed for this pressure and it would, therefore, be possible to use in PWR some of the equipment designs developed for those plants.

Selection of average reactor coolant temperature. To permit detail design of the steam generators, as well as to allow the core design to proceed, it was necessary early in the plant design to set the average coolant tempera-

ture at full reactor power. Before selecting this temperature, two other temperature limits were established, as previously mentioned. The plant specification set the steam temperature at 486°F (saturation temperature for 600 psi). The core hot spot temperature was set at 636°F after extrapolation of core material corrosion data. It was recognized that for a fixed steam temperature, steam generator heat transfer surface and cost decrease, but that core heat transfer surface and cost increase as the coolant temperature increases. However, because of the much higher cost of core heat transfer surface (compared with steam generator heat transfer surface), the average coolant temperature at full power was set while still maintaining a practical steam generator size. The average temperature so chosen was 523°F.

Selection of pumping power. Since PWR was to be an experimental plant, it was built to be flexible, able to accommodate many types of reactor cores. The quantity of reactor coolant flow that future cores might require was carefully considered. It appeared that although the first PWR core (Core I) would require a high flow at a relatively low pressure drop, future cores might require lower flows at a higher pressure drop. The pumps, as well as other equipment in the reactor coolant system, were designed to meet either condition. In the case of the pumps, it may be necessary to change impellers for any future high pressure drop cores, but this can be done with relative ease.

Pumping power was set at the estimated amount needed to provide the system flow required for the estimated maximum pressure drop. The possible economic advantage of increasing coolant flow rate was studied in an effort to reduce core heat transfer surface area and thereby reduce core costs. However, since pumping power increases as the third power of the coolant flow rate for any fixed system, it appeared that a total hydraulic power of about 1100 to 1200 hp per loop would be a good balance between the desire to decrease pumping power to increase plant efficiency and lower capital costs and the desire to increase pumping power to decrease core costs.

The pumps were designed to have two operating speeds to save pumping power and increase plant efficiency at reactor power levels below 50%. This appeared desirable because, as an experimental plant, PWR would probably operate more frequently at low power than would a conventional commercial power plant.

Selection of type of reactor reactivity control. A study was made of the feasibility of using chemical means, rather than mechanical, for core reactivity control of xenon override and fuel depletion. From the study it appeared that such control was feasible, but apparently not superior to mechanical rod control. The study also indicated that some rod control was desirable for safety shutdown and for providing large changes in

reactivity such as required in going from a cold to a hot plant. When the seed and blanket core design was adopted, work on chemical plant control was terminated, because even a small amount of control chemical in the reactor coolant would significantly reduce the amount of power obtainable from the natural uranium blanket. In addition, it appeared that it would be difficult to make a chemical control system fail-safe. Since the reactor control plays a major role in plant safety, this was considered a significant disadvantage.

After mechanical rod control was selected, a study was made to determine whether it was preferable for control to be based on constant average coolant temperature (T_{av}) or constant steam pressure. Constant T_{av} control would permit simpler reactor control, since the inherent negative temperature coefficient was expected to be high enough that no control rod motion would be necessary to compensate for normal load changes on the plant. It would, however, result in a large decrease in steam pressure in going from no load to full load. Accordingly, steam plant piping and equipment would have to be designed for a pressure of at least 850 psi (no-load pressure), although it would operate at lower pressures (600 psi) at full load. An estimate indicated that about $50,000 would be added to the over-all plant costs for constant T_{av} control. Since this type of control would result in more stable reactor plant operation, it was decided that this expenditure would be a good investment, and the design was firmed up on the basis of control of T_{av}.

1–5. Reactor Design Philosophy

1–5.1 Over-all objectives. In developing the reactor design, certain objectives were considered paramount. These are given below, although not necessarily in order of importance:

(1) That the core be designed so that it could be instrumented to measure actual fuel temperatures, coolant temperatures, and coolant flow at various stages of its life, for comparison with calculated results, since this would be of significant value to water cooled reactor technology.

(2) That the reactor incorporate a system to detect and locate failed natural uranium fuel elements.

(3) That the failure of a few fuel elements should not necessitate the immediate shutdown of the reactor; rather, that the reactor be capable of operating with a number of failed fuel elements for a considerable length of time (months) until these failed elements could be conveniently removed.

(4) That the failure of a fuel element must not cause adjacent fuel elements to fail also; furthermore, in case a fuel element fails and all of its uranium content is distributed throughout the reactor coolant systems, that the resulting radiation levels must still be within acceptable limits.

(5) That any fuel assembly be readily removable through ports in the pressure vessel head, thus minimizing the time necessary for removing an assembly containing a failed element or for actual refueling operations. (An additional requirement was that the core be removable as a cartridge, once the pressure vessel head was removed.)

(6) That the reactor design be such that fuel assemblies can be readily relocated in different positions of the core.

(7) That promising refueling concepts be developed and tried out; specifically, that the merits of wet versus dry refueling methods for fuel assemblies be compared by actual performance.

(8) That the reactor design be mechanically simple and contain a minimum of mechanical fasteners; furthermore, to retain design simplicity, that it incorporate only one pass of coolant through the core.

(9) That the reactor pressure vessel design make a significant contribution to the technology of designing and fabricating large high pressure reactor vessels.

(10) That the reactor be capable of producing its design power on the basis of proven thermal criteria — any advancements in such criteria could be taken advantage of later to increase core output.

(11) That the design power of the reactor be computed on the basis of operation at equilibrium xenon concentration.

(12) That the reactor be capable of shutdown with at least one control rod stuck in its uppermost position.

(13) That the reactor design be such that it can sustain the loss of one main coolant loop during rated full power operation without requiring a scram.

(14) That the fuel element design be suited to potentially low cost fabrication methods.

(15) That 50% or more of the core power be generated from the natural uranium in the core.

(16) That the core have a minimum life of 3000 full power hours.

From the basic PWR specifications and the above objectives, a core design was evolved. Some of the more important features that were finally chosen for the reactor core are discussed below.

1–5.2 Reactor core. *Thermal and hydraulic design objectives.* Thermal and hydraulic design of the PWR core was directed toward the previously stated objectives of 60,000 kw net electrical output or 225,000 kw thermal output from three reactor coolant loops, pumping requirements of approximately 1200 kw at normal operating temperature, 2000 psia operating pressure at reactor outlet, 600 psia steam pressure at the steam generators at full load, one-pass coolant flow through the reactor, and continued

operation without scram if one main coolant loop should fail during full power operation on either three or four loops.

However, the main criteria forming the basis for the thermal design of the PWR core were as follows:

(1) Local boiling was not to occur in the core under normal full load conditions, because of a lack of burnout data for flow geometries other than flow in single round tubes. To avoid local boiling with 2000-psia water, the maximum metal surface temperature of any of the fuel elements was limited to 636°F in the design. Inasmuch as normal errors in instrumentation could cause the plant to be operating at conditions other than those specified, the core performance was evaluated on the basis of assumed adverse errors of 30 psi in coolant system pressure and of 5°F in core inlet coolant temperature.

(2) To avoid any possibility of burnout, bulk boiling was not to occur in the water leaving the hottest channel during a loss-of-coolant-flow accident. During such a transient the bulk water temperature leaving the hot channel was permitted to rise to 636°F, the saturation temperature corresponding to 2000 psia.

(3) The maximum temperatures in the core components were not to exceed permissible values based on the mechanical and metallurgical properties of the material involved. This criterion did not impose a severe limitation on the design except insofar as temperature differences induce thermal stresses in the materials.

In addition to the above factors, the thermal and hydraulic operating characteristics of the PWR core were dependent upon the power plant parameters discussed previously and upon the detailed power distribution pattern and associated cooling within the core.

Mechanical design features. Core mechanical design features discussed earlier include physically interchangeable fuel assemblies, a minimum of mechanical fasteners, a fuel element design suited to potentially low-cost fabrication, and extensive instrumentation to yield performance information, to protect the core against thermal damage, and to monitor core operations. Other main features of the design were as follows:

(1) The PWR reactor is so designed that it can be refueled in one of three ways:

(a) By replacing individual fuel assemblies through fuel ports in the reactor vessel head, without removing the head.

(b) By replacing individual fuel assemblies after the head has been removed.

(c) By replacing the entire core cartridge, or core assembly, after the head has been removed.

The design is such that by completely removing the core components, a new and different core design can be installed.

(2) The core (and the reactor vessel) is capable of withstanding normal operating pressure surges of approximately ± 185 psi from the steady-state pressure. The duration of these surges can be from 1 to 5 min.

(3) The core is capable of sustaining a normal warm-up and cooldown rate of 200°F per hour between ambient room temperature and operating temperatures and, in addition, can withstand the following temperature changes:

(a) Increase of 50°F in 45 sec.

(b) Decrease of 30°F in 45 sec.

(4) Control rod drive mechanisms are designed to be compatible with temperature and pressure requirements outlined for the core and vessel. In addition, they are designed to minimize damage during a powerless emergency cooling operation of 36 hr duration with coolant temperatures in the pressure vessel in the region of the mechanisms at a maximum of 630°F. Each control rod has its own individual drive mechanism. Furthermore, each control rod can be removed directly through the mechanism port after taking out the mechanism which drives that rod.

Nuclear aspects of core design. The seed and blanket type core used in PWR represents an advanced physics concept that was aimed at achieving (1) the maximum percentage of power from the natural uranium in the blanket, (2) a minimum investment and expenditure of highly enriched fuel, (3) a minimum core size for a given power output, and (4) generation of plutonium and burning it in place in the core. There are a number of other important advantages to the seed and blanket core concept. Some of these, discussed more fully in Chapters 3 and 4, are:

(1) Reduced requirement for mechanical control.

(2) Large negative temperature coefficient of reactivity.

(3) Uranium enrichment not a variable.

Fuel and material design considerations. The design objectives for the fuel elements, described earlier in this chapter, limited to a large extent the possible fuel materials that could be used. A highly enriched uranium-zirconium alloy was chosen for the seed fuel mainly because experience indicated that this material could sufficiently resist radiation damage and corrosion in high temperature water if cladding failed. The same did not apply, however, to the blanket fuel element. Here, a fuel element containing as much natural uranium as possible was desired, and the material had to resist radiation damage at high burnups as well as resist corrosion in high temperature water. From a nuclear standpoint, a completely natural uranium fuel element would have been very desirable. However, such a material has extremely poor corrosion resistance in high temperature water and will distort upon extensive irradiation.

The fuel element development program was concentrated on two materials — uranium-molybdenum and uranium oxide. The uranium-

molybdenum fuel element as finally developed had satisfactory short time resistance to corrosion in high temperature water but would eventually fail by extensive disintegration. Thus, it was concluded that uranium-molybdenum fuel elements could seriously contaminate the coolant systems if cladding failed in a number of fuel elements. Uranium oxide was finally chosen for the blanket fuel material because tests indicated it was inert in high temperature water and had good resistance to irradiation damage when subjected to high burnup under PWR conditions. However, in case of a cladding failure, gaseous fission products would be liberated from the UO_2 fuel element into the coolant. Even so, tests indicated that such gaseous fission products would probably not activate the systems so much that they could not be satisfactorily decontaminated. The work in developing and fabricating these fuel elements is described in detail in Chapters 5 and 6.

Zircaloy 2 was chosen as the cladding material for both the seed and blanket fuel because of its low neutron absorption cross section, good corrosion resistance, and adequate strength in high temperature water; also, there was already a significant amount of operational experience with this material. It is interesting to note that in the process of developing the Zircaloy clad shape for the uranium oxide blanket material, a new industry had to be created where none existed previously — the manufacture of high-quality zirconium tubing.

The fuel element shapes and dimensions were based on a number of considerations including their thermal performance, the forms in which the material could be manufactured, and whether the mode of failure of an element could be such as to lead to the successive failure of adjacent elements. Some experience was already available with plate elements, and since such elements were thermally attractive, a plate geometry was selected for the uranium-zirconium seed elements.

At the time the blanket element design had to be decided upon, the only proven way of making high uranium content uranium oxide elements was by assembling them from pellets of the material; accordingly a rod-shaped element containing UO_2 pellets was chosen for the blanket. Some tests were performed to confirm that a failure of a UO_2 rod element would not cause its adjacent rod element to fail under PWR operating conditions. The diameter of the rod was based on a compromise of physics considerations, central fuel temperatures, surface area, and costs. The length of the rod was limited by the consideration that the failure of a rod element could possibly liberate all the uranium oxide contained in it into the coolant systems, which could present a serious contamination problem.

Hafnium was chosen for the control rod material since it had the desired neutron absorption characteristics, good corrosion resistance, and adequate physical properties in high temperature water, as well as satisfactory

resistance to radiation damage. Furthermore, significant experience in manufacturing this material was already available.

Throughout the core and pressure vessel, AISI type–304 stainless steel was selected as the principal structural material because of its excellent corrosion resistance, good fabricability, and high ductility, even under irradiation. The advantage of high ductility is that if a crack or other defect should develop, it is less likely to become serious enough to lead to a failure than is a similar flaw in a less ductile material. Even if the crack should, over a period of time, progress to the point of penetrating a pressure containing wall, the failure would develop slowly and would be detected by the steam leak that would result.

Primarily because of strength, wear, and galling considerations, the use of other special materials such as stellite and precipitation hardened stainless steel was found necessary in the control rod drive mechanisms.

1–6. Considerations in Design of Reactor Plant Facilities

1–6.1 Container and shield. At the initiation of the PWR Project it was not known where the plant would ultimately be located, and it was therefore necessary to proceed with the design on the basis that it would probably be located in the vicinity of a large center of population. In addition, it was considered desirable at the time to contain any type of reactor plant so located in a pressure-tight housing. Thus a decision was made prior to the start of the detailed design of the plant that the reactor plant would be contained in a pressure-tight container which would be located mostly underground. It is important to recognize, however, that this decision was not a result of a study which showed that such a container was necessary. Now that the plant design has been completed, the necessity for such a container can be questioned.

A study was made to determine the most economical container configuration: a single sphere, multiple spheres, a single cylinder, multiple cylinders, pressure tight concrete chambers, and cut and cover type structures consisting of a steel container with earth shielding. It was found that the most economical shape for location underground with external shielding is a cylinder. The final form of the container is that of three cylinders and a sphere, interconnected by large diameter piping.

The container was sized to contain the expanding steam from the release of all hot reactor coolant in the event of a reactor coolant system rupture plus the simultaneous rupture of a steam generator. It was assumed that a major rupture in a steam generator might result not only in release of the energy stored in the secondary side water, but in a rupture of the reactor coolant system as well. (It did not appear reasonable to assume that there would be a simultaneous rupture of the reactor coolant system and more

than one steam generator.) The container design pressure was set at 52.8 psi based on the diameter of the cylindrical section required to house the equipment and the maximum steel thickness that could be welded without stress relief. The container was pneumatically tested at 70 psi.

One factor that affected the design of the containers and controlled the design of the internal shield was the requirement that the plant would have to be maintained and repaired while in operation. Therefore, separate shielding was provided for each loop, and many auxiliaries were located in separate shielded access areas in the container, outside the loop areas. Pressure-tight air locks were provided for personnel access to these loops and access areas while the plant was operating. There is, however, no necessity for anyone to enter the container for routine operations or inspections.

The exact configuration of the container resulted from several basic requirements. As mentioned previously, it was considered highly desirable, from the standpoint of plant safety, to locate the reactor vessel at an elevation lower than that of the steam generators to permit natural convective circulation of reactor coolant between these two components. This feature provides a means for cooling the core in case all power to the Station should ever be lost. To provide this difference in elevation, it proved desirable to separate the reactor chamber from the rest of the container. To conserve space at the site in a direction parallel to the river so as to provide room for installing future power plants, the boiler chambers were made as short as feasible and the additional container volume required was provided by an auxiliary chamber located adjacent to the boiler chambers. Another factor that influenced the general configuration of the container and shielding was the requirement for underwater refueling. In order not to have an excessively high fuel handling building and still provide all the water depth required for shielding active cores, it was found economical to locate the top of the fuel canal near ground level and arrange it so that direct access from the canal to the space over the reactor vessel was possible.

1–6.2 Fuel-handling facilities. Experience in the design of previous reactors had shown that the cost of providing a shielded coffin to be used in refueling an entire activated core as a unit would be costly, if not impractical, for a core as large as that of PWR. However, to save time during refueling and to give the plant as much flexibility as possible for future core designs and experimental work, it was decided to provide for refueling the core as a unit. The least expensive way of doing this proved to be a water filled canal so arranged that it has direct access to the reactor vessel. In addition, the experimental nature of the plant indicated that being able to insert and remove experimental core fuel assemblies with a minimum

of plant outage time would be highly desirable. This was made possible by providing the necessary remote handling equipment and designing the core and reactor vessel to permit removing and inserting any core subassembly without taking off the reactor vessel head and all the equipment mounted thereon. Two methods for removing core subassemblies were developed: one involved removal under water; the other, removal with a shielded container.

To provide adequate storage space for activated cores and to reduce the number of expended fuel shipping containers, it was decided to provide underwater storage facilities for two completely activated cores. Since any PWR core is expected to contain a large number of separate assemblies, it was not considered economical to provide enough shipping containers so that all these subassemblies could be shipped from the site in a short period. Rather, only a small number of subassemblies will be shipped at one time, and the containers will be returned for subsequent loads. Several months will thus be required to move an expended core from Shippingport to its processing point. During this period, it may conceivably be necessary to remove an active core from the vessel before the previous core has been completely shipped from the site. This is why the core storage facilities must be capable of handling two expended cores at one time.

Experience with previous plants had shown that valuable information can be obtained from visually examining core surfaces immediately after they are removed from the reactor, before they are shipped to a reprocessing center. It was, therefore, decided to supply a periscope and underwater camera television equipment for examining expended core parts in the fuel handling canal. It was not considered desirable, however, to provide "hot" laboratory facilities for direct viewing of core components, or to provide for cutting open of fuel elements with the resultant escape of fission products into the area. Such experimental work would be performed at locations properly equipped with hot laboratory facilities for such specialized work.

Storage facilities for one complete new core were provided at Shippingport to help minimize plant down time during refueling and to allow for the possibility that not all future core suppliers will be able to store completed new cores on their premises. The new core vault was built adjacent to the fuel-handling building. Since it may not be desirable to ship a new core completely assembled, a core assembly room was also provided adjacent to the fuel-handling building.

1–6.3 Service facilities. In general, service facilities at Shippingport have been kept to a minimum consistent with the experimental nature of the plant and local operating conditions. The chemistry laboratory facilities

are somewhat more extensive than will probably be necessary in future conventional reactor power stations. However, reactor water chemistry is one of the areas in which reactor technology is still in an early stage of development; these laboratory facilities are expected to be very useful in adding to the existing knowledge in this area, as well as in that of radioactive waste disposal. A laundry for radioactive clothing has been provided as a result of a study which indicated that it might be more economical to launder contaminated clothes at Shippingport, where adequate waste disposal facilities exist, than to have it done commercially. A decontamination room with a stainless steel floor is provided for decontaminating equipment that is too radioactive to be repaired at conventional machine shops, most of which are not equipped to handle such equipment. The remaining facilities in the service building, such as the shower rooms, health physics facilities, tool rooms, and storage areas, are considered the minimum essential for this plant. A second floor can be added to the service building should future experience indicate that additional space is required.

1–6.4 Plant parameters. Major plant parameters are given in the Appendix. Reactor coolant system parameters will be found in Chapter 2, reactor parameters in Chapter 4, and radiation level specifications in Chapter 13.

CHAPTER 2

REACTOR COOLANT SYSTEM

CHAPTER 2

REACTOR COOLANT SYSTEM*

Heat is carried from the reactor core to steam generators by the reactor coolant system. Highly purified water circulates through three coolant (radioactive) loops to the steam generators, carrying reactor heat which is transferred to secondary (nonradioactive) loops, and forms steam which drives the main turbine. A fourth coolant loop is provided as a spare. The performance data of other reactor plant systems which affect the reactor coolant system are given in Chapter 8.

2–1. Summary Description of System

The reactor coolant system consists of four closed piping loops (one is a spare) connected in parallel to the (single) reactor vessel. Each coolant loop contains two inlet stop valves, a steam generator, a coolant circulating pump, a check valve, a venturi flowmeter, temperature and pressure instrumentation, and two outlet stop valves. Each loop also has a bypass line connecting the inlet piping to the outlet piping, bypassing the reactor vessel. A schematic diagram of the reactor coolant system is presented in Fig. 2–1.

Each of the four coolant loops contains a steam generator for use as a heat exchanger. Two of the loops use a straight-tube heat exchanger manufactured by the Foster Wheeler Corporation; the other two use a U-bend heat exchanger manufactured by the Babcock & Wilcox Company. Because of this difference in steam generator design, two different loop arrangements were used. Experience gained in the design, manufacture, and operation of both types will be helpful in future reactor plant design.

Each coolant loop is located within its own shielded compartment; the four compartments are located around a central compartment containing the reactor vessel. The shielding around each loop compartment is designed to permit personnel to enter the compartment of a shutdown loop while the other loops are operating at full power output.

2–2. System Design Requirements

The net electrical output (according to design specifications) of the Shippingport Atomic Power Station is 60,000 kw. The three-loop reactor

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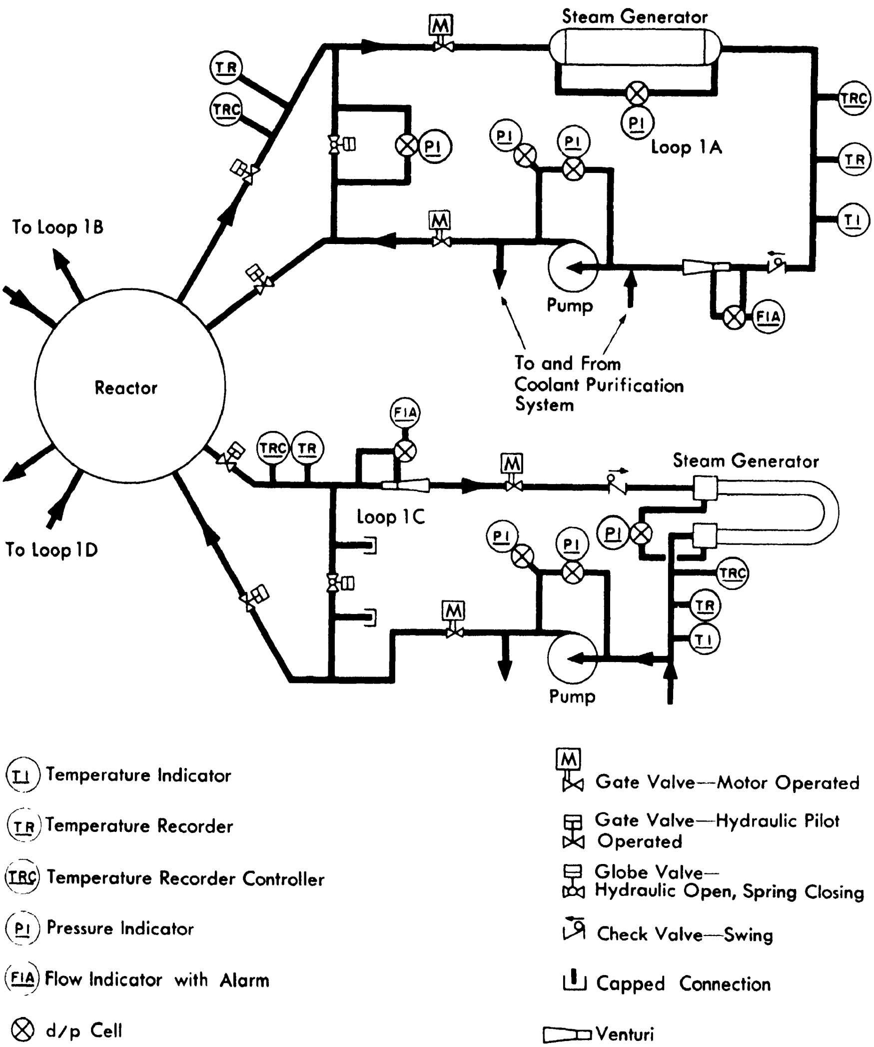

FIG. 2-1. Reactor coolant system.

flow rate is approximately 22.8×10^6 lb/hr; with four-loop operation, the flow rate is approximately 28.8×10^6 lb/hr and a somewhat greater net electrical output can be realized. In normal three-loop operation, the heat transfer load from the reactor is 768 million Btu/hr (225,000 thermal kw), and the gross electrical output is about 68,000 kw. Service requirements for the station are about 5000 kw for the nuclear portion of the station and 3000 kw for the turbine-generator portion; gross electrical output of 68,000 kw is therefore required for the 60,000 kw net design output.

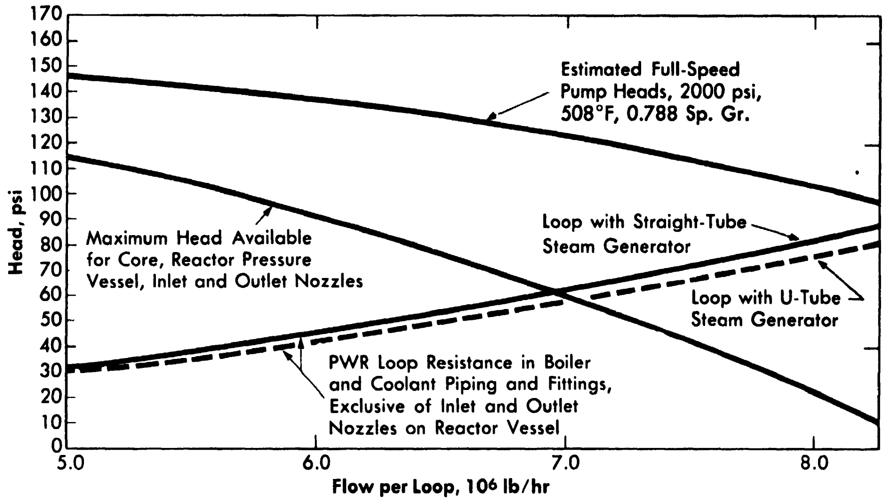

FIG. 2–2. Head and flow characteristics.

The normal operating pressure at the outlet of the reactor vessel is 2000 psia. The system and its components are designed to handle pressures of 2500 psig. With the present core (Core I) installed, the reactor coolant system operates with a reactor inlet temperature of 508°F and a reactor outlet temperature of 538°F. Thus, the average coolant temperature is 523°F. The design temperature of the system and its components is 600°F.

The system was designed to accommodate not only Core I, but also a variety of other core types which are being considered for future installation. The head-flow characteristics of these future cores vary over a wide range, and the system has been designed so that only the impellers of the pumps need be changed to satisfy varying requirements. Figure 2–2 shows the full speed head-flow curve for the coolant pumps, the resistance in the loop piping and components, and the head available for the reactor core and reactor pressure vessel as a function of the loop flow rate. The latter curve satisfies the requirements of all cores now under consideration for future installation in PWR.

The reactor coolant system is all-welded construction; system components are seal-welded to keep coolant from leaking to the plant container. All main flowstream components of the system are designed to withstand the following coolant temperature transients or their equivalent:

(1) 400 warmup and cool-down cycles between ambient (room) temperature and full operating temperature at the rate of 200°F/hr.

(2) Any one set of the following rapid transients resulting from load changes:

	*Transient**	*Number of transients*
	$\pm 6, \quad \mp 3°F$	200,000
or		
	$\pm 12, \quad \mp 6°F$	100,000
or		
	$\pm 16, \quad \mp 8°F$	25,000
or		
	$\pm 20, \quad \mp 10°F$	7,000

The steam generators are designed to withstand any one set of the following secondary water temperature changes while at operating temperature:

*Transient**	*Number of transients*
(1) $\pm 5, \quad \mp 2°F$	200,000
(2) $\pm 9, \quad \mp 3°F$	100,000
(3) $\pm 14, \quad \mp 5°F$	25,000
(4) $\pm 18, \quad \mp 7°F$	7,000

The system was designed in accordance with the requirements of the Department of Labor and Industry, Commonwealth of Pennsylvania, whose regulations govern the design of power plants erected within the Commonwealth.

2-3. Detailed Description of System

2-3.1 Main stop valves. Each loop contains four main stop valves, two in the inlet and two in the outlet piping of the reactor.† One stop valve in the inlet and one in the outlet line are inside the reactor chamber close to the reactor vessel. These can be hydraulically operated by remote control from the main control console to isolate a given loop from the reactor vessel. The remaining stop valves in each loop are inside the boiler compartments, one in each of the inlet and outlet lines of the reactor.

* The first part of each temperature transient occurs in approximately 50 sec and is followed by the second transient approximately 100 sec later.

† Paragraph P-303 of Section I of the ASME Boiler and Pressure Vessel Code (the code adopted by the Pennsylvania Department of Labor and Industry) was considered applicable to the loops in the PWR reactor coolant system. This paragraph requires two stop valves, with an "ample free-blow drain between them," in a steam connection from a boiler to a steam main, if the boiler has a manhole.

These valves have their topwork completely covered with leaktight caps during normal operation and are designed to work either manually or by electric motor after the caps are removed.

When maintenance on a loop is required during plant operation, the remotely controlled hydraulic valves are closed and the loop is drained. The boiler compartment is entered, the cap is removed from the locally controlled valve, a motor or handwheel is installed, and the valve is closed. Telltale (free blow) valves between the stop valves are then opened and maintenance of the loop is begun.

All main stop valves have body connections suitable for butt welding to 18-in. nominal size pipe, but have 16-in. nominal size throat diameters. Throat sizes are increased gradually to ensure smooth flow. The maximum water velocity at the throat is approximately 46 fps.

The hydraulically operated valves are installed with their cylinders oriented below the valve body so that the gate and piston assembly will not drift closed due to gravity. No leakage to ambient is permitted; leakage between the operating cylinder and the valve body is permitted when the valve gate is closed. A back seat limits leakage from the valve body to the operating cylinder when the valve gate is open.

Position indicators are provided for each valve to indicate when the valve has reached the fully closed and fully open positions.

2–3.2 Reactor coolant pumps. One vertical single-stage centrifugal pump is installed in each of the four reactor coolant loops. The impeller is attached to the rotor shaft of an induction motor, with the stator and rotor encased in corrosion-resistant cans designed to withstand full system pressure. Cooling water of the same purity as coolant water, but at much lower temperature, circulates between the stator and the rotor, lubricates the motor bearings, and eliminates the need for a seal around the motor shaft. The motor is seal-welded and bolted to the pump casing, making a completely leakproof unit.

To conserve pump power at low plant electrical output, the pumps are two-speed (full and half) units driven by 2300-volt motors. Characteristic curves for full- and half-speed operation are shown in Figs. 2–3 and 2–4, respectively.

These pumps can operate at low-flow and high-head or high-flow and low-head conditions.

Required flow	*Total developed head*	*Water temperature*
4.0×10^6 lb/hr (10,000 gpm)	145 psi	501°F
4.78×10^6 lb/hr (12,000 gpm)	136 psi	506°F
7.62×10^6 lb/hr (19,300 gpm)	111 psi	508°F

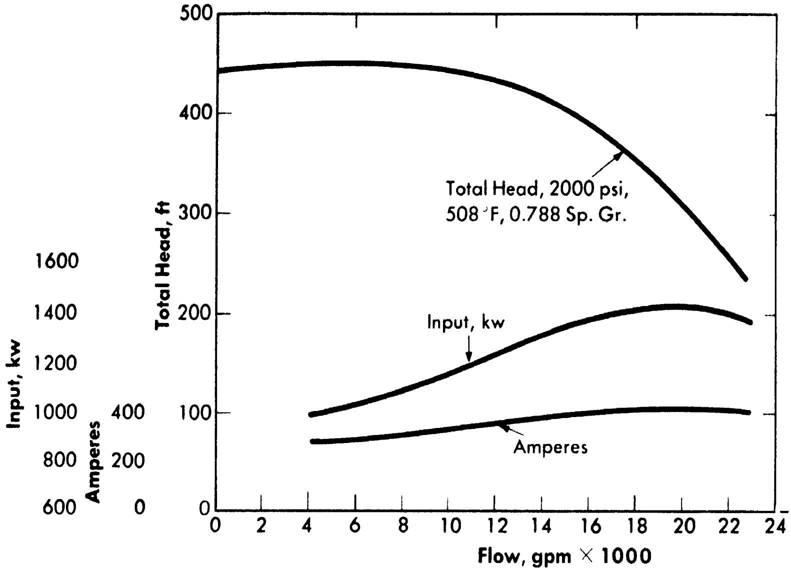

FIG. 2–3. Main coolant pump characteristic curves for full-speed operation.

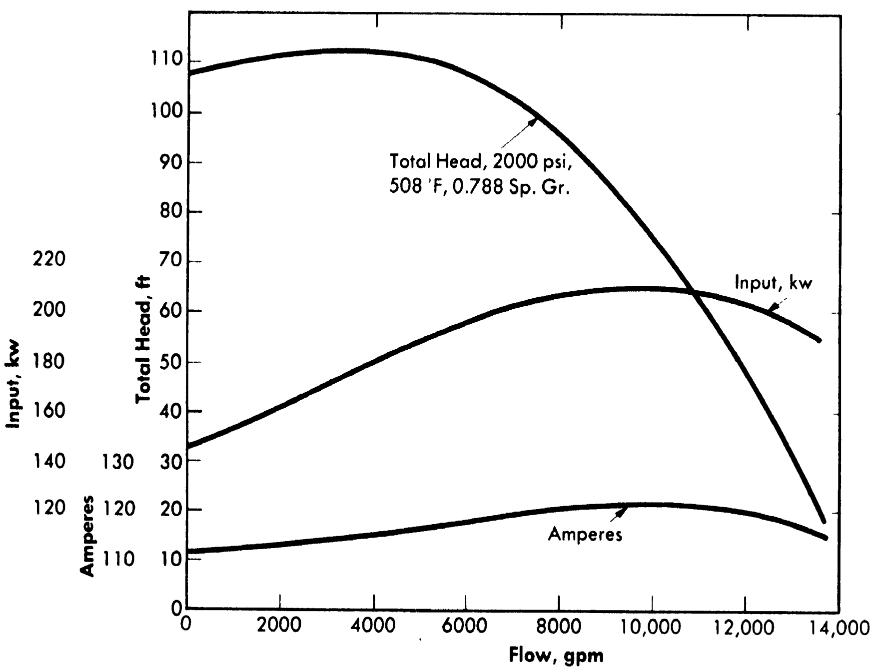

FIG. 2–4. Main coolant pump characteristic curves for half-speed operation.

These capacities are quoted for full-speed operation. The pumps are capable of meeting the requirements of the conditions above with the same volute, but impellers of different diameter must be installed to obtain these two extremes of design operating points. The pumps have cast type-304 stainless steel volutes and impellers, and forged type-304 stainless steel pressure shells, flanges, and rotor shafts.

2-3.3 Steam generators. The steam generators are shell and tube type units in which reactor coolant inside the tubes heats secondary water in the shell. Both straight and U-bend generators were used to gain design, manufacturing, and operational experience that will be helpful in the design of future plants. Loops 1A and 1D have Foster Wheeler Corporation generators of the straight tube design. Loops 1B and 1C have Babcock & Wilcox Company generators of the U-shell design.

The two straight tube heat exchangers each contain 2096 stainless steel tubes, 31 ft long with ½-in. outside diameter. These tubes are rolled and welded into a stainless steel tube sheet and are enclosed by a stainless steel shell 43 in. in diameter. The ends of the heat exchanger portion of the unit have hemispherical heads of stainless steel with 18-in. pipe connections. The conventional steam drum (boiler), fabricated entirely of carbon steel, contains 48 separators and 20 driers to reduce moisture content of the steam. Six 8-in. downcomers and twelve 8-in. risers connect the heat exchanger to the steam drum.

The two U-bend heat exchangers each contain 921 stainless steel tubes having an average length of 50 ft and an outside diameter of ¾ in. Rolled and welded into stainless-clad carbon-steel tube sheets, these are enclosed by a U-shaped shell 38 in. in diameter. The ends of the heat-exchanger portion of the unit have hemispherical carbon-steel heads. Internal surfaces of these heads, as well as the face of the tube sheet in contact with coolant water, are clad with stainless steel. The conventional steam drum is constructed of carbon steel. It contains 30 steam separators and two rows of secondary scrubber elements, which separate water particles from the steam. Fourteen 4-in. downcomers and eighteen 5-in. risers connect the steam drum with the heat exchanger.

Three units are capable of producing the steam required to generate 60,000 kw net electrical plant output. The steam generator operating conditions for Core I are specified as follows:

Heat load per unit: 256×10^6 Btu/hr.
Reactor coolant flow rate per unit: 7.62×10^6 lb/hr.
Reactor coolant temperature entering steam generators: 538°F.
Reactor coolant temperature leaving steam generators: 508°F.
Steam pressure at drum outlet: 600 psia, dry and saturated.
Maximum allowable reactor coolant pressure drop: 42 psi.

The steam generators are designed to function satisfactorily under either constant or variable heat load. Both of these operating conditions may include either high or low coolant flow. The steam generators are designed for a constant heat load of 263×10^6 Btu/hr each. The design variable heat load is 250×10^6 Btu/hr. Design parameters include:

	Constant heat load, high flow	*Constant heat load, low flow*	*Variable heat load, high flow*
Coolant flow rate, lb/hr	7.62×10^6	4.60×10^6	6.28×10^6
Coolant inlet temperature	538°F	554°F	541°F
Coolant outlet temperature	508°F	506°F	508°F
Maximum coolant pressure drop, inlet nozzle to outlet nozzle, psi	35	19	29

The secondary sides of the steam generators are designed for a pressure of 1200 psia, with a maximum no-load steam pressure of 885 psia for both operating conditions.

2-3.4 Check valves. Each loop has one swing check valve to prevent excessive backflow if the pump stops. If the plant is operating at 60,000 kw net power output with three loops in service (the reference design condition) and one coolant pump stops, the reactor loses only the flow normally produced by the stopped pump; if no check valve is installed, backflow through the stopped pump would cause additional loss of flow approximately equal to the output of one of the remaining pumps. Depending on the design characteristics of the core installed, such a loss of flow might cause scramming of the reactor. Furthermore, reverse flows of the order of 3000 to 6000 gpm for prolonged periods might damage the pump bearings.

The reactor coolant loops must permit low head thermal circulation during removal of reactor decay heat. Since the loop check valves have low resistance to normal flow, they are designed so that they will not close unless there is a backflow of 1000 gpm. To provide more flexibility in operation, a 13/16-in. diameter hole has been provided in the disk of each check valve, permitting a small reverse flow in a coolant loop that is not operating but not isolated, thereby providing a means for maintaining the temperature in that loop at the operating temperature of the plant. This avoids cold water reactivity accidents that might be caused by accidental pump starting.

Theoretically, the check valves can be located at any point in the coolant loops. Since the physical arrangement of the piping made it impossible to

place the check valves in similar points in all loops, the 1A and 1D loop check valves are located in the cold-leg piping on the suction side of the pumps, and the 1B and 1C check valves are located in the hot-leg piping upstream of the steam generators.

2-3.5 Bypass lines and valves. Each loop contains a valved bypass line which is connected to the main piping on the loop sides of the hydraulic stop valves. These lines extend into the individual boiler compartments, so that their valves are accessible for maintenance. The loop telltale drain valves required by the ASME Boiler Code are also operable from the boiler compartments. They are provided with hoses and can be used to drain nonradioactive reactor coolant water (at less than 150°F) to the reactor plant gravity drainage system through low-point floor drains provided throughout the plant.

With the bypass valve open and the main hydraulic valves closed, the coolant pump will circulate water through the bypass line to warm up a loop containing cold water. The interloop flow rate required for warmup is approximately 100 gpm. The coolant pump is operated at half speed when the loop is isolated and the bypass valve is open.

2-3.6 Main piping. The main piping for the system is 18-in. nominal OD by 15-in. nominal ID. The pipe design is based on paragraph P-23 of Section I of the ASME Boiler and Pressure Vessel Code. For 18-in. OD seamless pipe with a zero corrosion and erosion allowance, the minimum pipe thickness, as determined by P-23, is 1.415 in.

The piping is hollow forged, turned, and bored to a finish of 125 microinches root mean square (rms) on the inside and 250 microinches rms on the outside; both finishes are well within currently available commercial tolerances. Fittings are made of the same material as the piping, and are also hollow forged.

2-3.7 Piping layout considerations. In general, measurements in connection with reactor operation showed that the areas of greatest radioactive contamination coincided with those that would naturally be suspected as particle traps. Consequently, piping layouts and components for all systems exposed to reactor coolant were carefully examined to determine which points might trap particles of fuel and crud. Several results of this review were as follows:

(1) Internal surfaces of piping and components were made as free from macroscopic irregularities and flow obstructions as possible. Latent trouble spots such as improperly drawn-up welding rings, weld spatter, and cracks and crevices were eliminated. (The microscopic smoothness of the surface was considered far less important than larger-scale irregularities.)

(2) All taps and penetrations for static or low velocity connections to piping in which the reactor coolant flows were located in the upper quadrants of the pipe no more than 45 degrees from the vertical. (Drain lines were the only exception.)

(3) Piping layout was designed so that drain lines or other connections which cannot satisfy requirement (2) above could be blown out to remove accumulated radioactive particles. Vigorous high-velocity flushing can be sustained long enough to sweep the material at least as far as the main connecting line to the radioactive waste disposal system.

(4) Velocities not less than 4 fps are required for quantitative transport of uranium oxide particles (micron size) through straight horizontal runs of piping with no obstructions, tees, or other flow disturbances. Where possible, piping which handles reactor coolant was sized to provide this minimum velocity. Where this could not be done, either arrangements were made for intermittent, high-velocity flushing, or local shielding was provided (in those areas where access may be required).

(5) Wherever possible, thermocouple wells were installed in a vertical position from the top of the pipes.

(6) In the coolant purification system, where possible, components were located downstream of the demineralizer.

(7) Sharp bends and unnecessary abrupt changes in the flow pattern, such as changes in pipe diameter, orifices, etc., were avoided. Where possible, venturis were used in place of orifices. The use of tee-connections in place of elbows was avoided except for those instances where the dead leg was directed upward or could be flushed out.

(8) Instruments such as plant pressure indicators which do not need to be located directly on the main loop were located on intermittently purged lines.

(9) When cracks and crevices could not be completely eliminated from components, they were opened up so that the width-to-depth ratios were as great as possible. A ratio of at least 1 to 4 was considered desirable.

(10) Where abrupt flow constrictions were necessary, every effort was made to locate these in vertical runs of piping.

As a result of this examination, particle traps were minimized in those systems handling reactor coolant.

2–3.8 Instruments, controls, and protective devices. *Operating instruments.* (a) Thermometers. Each coolant loop contains two thermometers in the hot-leg piping and three in the cold-leg piping. The hot-leg thermometers penetrate the 18-in. pipe just downstream of the reactor outlet hydraulic stop valve. The cold-leg thermometers penetrate the pipe just downstream of the steam generator outlet nozzles.

All thermometers are resistance units capable of operating over a tem-

perature range from 50 to 650°F. Sixteen of these units are high-speed elements (two to three seconds for 63% response to a step change) and are welded directly into the piping. These elements are normally used only in the range of 450 to 650°F. Four units are slow-speed platinum resistance thermometers designed for high accuracy (less than ±0.25°F error over the operating range). These elements are mounted in separate wells in the cold-leg piping and are used to calibrate the high-speed elements.

A pair of high-speed resistance thermometers is mounted at each reactor outlet and a pair is installed at each steam generator outlet. One of each pair is used as an indicating and control thermometer; the second of the pair in the cold leg is used in the temperature difference interlock circuitry, and the second of the pair in the hot leg is used for wide-range temperature indication and as a spare. All temperatures are indicated at the main control room.

One high-speed resistance thermometer is located in the secondary side of each steam generator to provide an input signal to the temperature difference interlock discussed later in Article 2–3.8.

(b) Reactor vessel differential pressure transmitters. Two reactor vessel differential pressure transmitters are installed, one in loop 1A and one in loop 1B. These transmitters are identical units, one backing up the other. They transmit reactor vessel differential pressure to the reactor section of the main instrument panel. This reactor vessel pressure drop indication is not used in the plant control system, but rather provides information useful to the plant operators and valuable to those analyzing the fluid dynamics of the reactor coolant system. These units are installed so that the connections penetrate the loop bypass lines, thus making it possible to remotely isolate the transmitters in the event of a leak.

(c) Coolant pump total head transmitters. Differential pressure transmitters are installed across each coolant pump to transmit the pump total head to the reactor section of the main instrument panel. This information, like that provided by the reactor vessel differential pressure transmitters, is used by plant operators and by engineers analyzing the system.

(d) Steam generator differential pressure transmitters. Differential pressure transmitters are installed at the steam generators and read out on the reactor section of the main instrument panel. This information, likewise, is not required for plant operation but is provided for plant analysis.

(e) Flowmeters. The reactor coolant flow of each loop is obtained by measuring the pressure drop across a calibrated venturi section in the reactor coolant piping with a differential pressure transmitter. The differential pressure cells are located about 50 ft from the venturis in an access cubicle, to facilitate venting and adjusting the differential pressure cells in an area of low radiation level.

Each differential pressure cell transmits an electrical signal to its associated flow indicator on the reactor section of the main control console. The range of the flow indicator is 0 to 23,200 gpm or 9.0×10^6 lb/hr of water at 2000 psia and 525°F. The venturi calibration is based on a temperature of 555°F to permit all the correction factors to be positive numbers for water at various loop temperatures. The calibrated accuracy for adjusted flow measurement is $\pm 2\%$ of full scale with a reproducibility of $\pm 1\%$.

Each recorder has four auxiliary contacts which can be set to open or close within $\pm 2\%$ of any preset flow rate over the full range. The auxiliary contacts are used as follows: one for low-flow alarm when the pump speed switch is set to full speed; one for low-flow alarm when the pump speed switch is set to half speed; and two spares which may be used for low-flow signals to the safety shutdown system, or for future needs of the control systems.

(f) Steam generator leak detectors. Four channels are provided in the operational radiation monitoring system for monitoring the secondary water in the four heat exchangers. Significant levels of radioactivity indicate leakage from the reactor coolant system to the main steam system. Because of the high radiation background within the boiler chambers, it was necessary to run sampling lines from the blowdown lines of the heat exchangers and from the steam drums through the shield walls. Cooling coils were provided on the sampling rack to lower the temperature of these monitor samples. A gamma-sensitive Geiger tube was attached to each line downstream of the cooling coils. Samples from each boiler pass through ¾-in. lines to the detector units at a flow rate of ¼ gpm. Each detector is connected to a panel-mounted count rate meter unit on the operational radiation monitoring system panel. The panel-mounted units convert the pulses received from the detectors to meter deflections.

(g) Instrument valving. The reactor vessel, steam generators, venturis, and coolant pump differential pressure transmitters are provided with single isolation valves in lines connecting them to the system; there are equalizing valves in the lines across the units. The venturi differential pressure transmitters, located in separately shielded cubicles, are accessible during system operation. A vent gas container with necessary valving is provided to contain gaseous fission products which may be entrapped at bleed-off points and released during the periodic purging of the cells. All purging is done with low system pressure (below 500 psi). For simplicity, the instrument valving has been omitted from Fig. 2–1.

Controls. The controls of the reactor coolant system must operate the coolant pumps and the hydraulic valves in the system. The controls, located on the reactor section of the main control console, actuate the breakers and speed selector disconnecting switches in the pumps' con-

trol circuitry (discussed below). They also position the system hydraulic valves.

(a) Pump controls. Pump motors are reconnected from half to full speed by motor-driven, electrically and mechanically interlocked, gang-operated disconnecting switches. The speed selector switches can be operated only when the associated coolant pump is off. After the pump speed has been selected, the associated "ON-OFF" pump circuit breaker control switch may be positioned to "ON" to energize the pump motor.

(b) Valve control. Both pairs of main-loop hydraulic stop valves in each loop are opened and closed by a position control switch on the console. This switch actuates a selector valve to direct the valve operating water to the proper side of the valve cylinders.

Protective devices. (a) Temperature difference interlock. The loop temperature difference interlock equipment gives a permissive control signal for coolant loop pumps of a down loop to operate if water temperature in the down loop is such that it will not cause a cold water accident in the reactor. A temperature differential of less than 20°F between the secondary water in the heat exchanger of the loop being placed in service and the highest "auctioneered" cold-leg temperature of the operating loops is required before the permissive signal will appear on the main and auxiliary instrument panels.

(b) Valve and pump interlocks. In addition to the cold water accident protection provided by the temperature interlock discussed above, the reactor coolant loop main hydraulic stop valves are also used to prevent such an accident: they are interlocked so that they cannot be opened unless the reactor control rods are fully inserted into the core. Shutdown signals are also provided to the reactor protection system if one valve moves from the open position while the plant is operating at more than 44% power on three loops. Interlocks are also provided to prevent operation of (and subsequent damage to) the coolant pumps at, or close to, their shutoff head. A coolant pump cannot be started on fast speed if the main stop valves in the loop are closed, and it cannot be started on slow speed if both the loop stop valves and the loop bypass valves are closed. Likewise, closure of a pair of loop stop valves will automatically shut down the pump in the loop if it is operating at a fast speed.

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CHAPTER 3

PHYSICS

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3-1. Introduction

The PWR core embodies the seed and blanket principle, an advanced and unconventional physics design for the nuclear generation of power. The chief advantages of this design are believed to be in the following areas:

(1) Reduced investment of enriched fissionable material.

(2) Generation of a considerable portion of the core power from natural uranium.

(3) Large negative temperature coefficient of reactivity.

(4) Reduced requirements for mechanical control.

Physically, the seed and blanket core consists of a relatively small, annular shaped "seed" region containing highly enriched fuel assemblies, and a "blanket" region of natural uranium fuel rods arranged in a lattice surrounding the seed regions both inside and outside. It is often helpful to think of the seed and the blanket as two essentially separate reactors. The seed is self-sustaining, but the blanket is somewhat subcritical (multiplication factor less than unity). The blanket thus operates at a "deficit" in each neutron cycle and, if it were not for the seed, would quickly stop producing power. However, neutrons leaking out of the seed into the blanket compensate for the "deficit" and enable the blanket to produce more than half of the core power. The seed may be said to "drive" the blanket. Control rods are thus required only in the seed.

Before entering into a detailed discussion of the physics characteristics of a seed and blanket core, a history of the PWR physics development program will be given to illustrate the main features and problems in designing a core of this type.

3-2. Historical Background

The idea of a seed and blanket core, as originated by Dr. A. Radkowsky, Senior Physicist of the Naval Reactors Branch, Reactor Development Division, U. S. Atomic Energy Commission in 1953, embodied the following:

(1) A core consisting of a small region of highly enriched uranium surrounded by natural uranium fuel elements.

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(2) A power distribution such that more than half of the total power output could be obtained from the natural uranium. To obtain as much power as possible from the natural uranium it would be necessary:

(a) To make the multiplication factor of the natural uranium lattice as high as possible.

(b) To arrange the seed in such a configuration as to cause as many neutrons as possible to leak to the natural uranium blanket.

(3) An initial blanket conversion ratio larger than unity, thus making it probable that the blanket would have a long reactivity lifetime. In other words, destruction of a given number of U^{235} atoms in the blanket would result in the formation of a larger number of plutonium atoms, each more reactive than a U^{235} atom.

(4) A core that could be refueled by replacing only the relatively small seed volume.

(5) A design that would have several schedular advantages. First of all, the seed region, which primarily determines the reactivity of the core, could to a large extent utilize the already developed technology of highly enriched uranium cores. Second, the design and manufacture of the blanket could begin without awaiting the selection of an enrichment value. In contrast, for a slightly enriched core design it would be necessary to complete a major portion of the physics work in order to select the enrichment. Third, since the seed and blanket design can use highly enriched and natural uranium fuels, the long lead time necessary to procure uranium of an intermediate enrichment would be avoided.

The subsequent development of this core concept was the product of close collaboration between the Naval Reactors Branch and the Westinghouse Bettis Plant.

In the initial nuclear design studies for PWR, a central seed location was investigated, but these exploratory studies showed that the power densities in a centrally located seed would be excessive. Although the seed power density could be reduced by increasing the radius of the seed, the result would be a reduction in the blanket power fraction because of the lower leakage of neutrons out of the seed. This may be understood if we picture the seed as a core in itself and the blanket as a reflector providing an essentially constant reflector saving to the seed. Another disadvantage of the central seed arrangement is the high radial power peaking factor* in the blanket. Since the blanket is subcritical, there is a tendency for the power level to decrease exponentially with distance from the seed, the rate of decay increasing as the blanket multiplication factor is reduced.

These difficulties were largely overcome by utilizing a seed in the form of a cylindrical annulus. With this geometry it is feasible to obtain a

* Peaking factor is the ratio of the maximum value to the average value.

relatively high neutron leakage from the seed while keeping the seed volume large enough to give reasonable seed power densities. At the same time, the blanket power distribution is improved since, with the annular seed shape, the average distance of the blanket fuel elements from the seed is reduced.

While the heat-transfer problems in the seed and blanket core at first seemed formidable because of the high power density in the seed, further developments indicated several features of this design tending to reduce the core volume required for a given power output, as compared with that of a uniformly enriched core. Since resonance capture of U^{238} is negligible in the seed, it can utilize thin fuel plates with a high surface-to-volume ratio, and therefore is well adapted to deliver a large fraction of the core power in a small volume. Also, as disclosed by life history studies, the shift in power between seed and blanket regions over a seed lifetime is relatively small, so that it is possible to orifice the coolant flow to each region so as to utilize the available coolant to maximum advantage.

From the early studies it soon became apparent that the seed and blanket reactor had the desirable property of a large negative temperature coefficient of reactivity. This results from reduction in water density with increasing temperature and is characteristic of highly enriched cores with high leakage. Such behavior of the temperature coefficient is a manifestation of the fact that the reactivity is determined principally by the properties of the seed. This was borne out by the fact that large variations in the various parameters of the blanket, such as water-to-metal ratio and fuel rod size, produced little effect on the core reactivity. A large negative temperature coefficient is difficult to obtain in a large core of uniform or nearly uniform fuel loading, because of the low leakage of fast neutrons from the core.

In December 1954, sufficient information was available to establish a reference design configuration. To provide high flexibility, a design was adopted utilizing an annular seed with additional inset seed assemblies, with the thought that these assemblies could be shifted to provide a change in the average annular thickness, if such a change were found necessary. The reactor was to be controlled by 32 hafnium control rods located in the seed, one rod per seed cluster.

After the reference core configuration was adopted, a detailed examination of the power distribution was begun. It was found that the power distribution was sensitive to the method of control rod programming. Originally, for simplicity, it was intended to move all 32 of the control rods as a bank, until it was found that extremely high axial peaking factors (in the neighborhood of 3 or 4 to 1) were obtained because of the crowding of the flux and power into the lower core region below the ab-

sorbing control rods. The explanation goes back to the model in which the reactivity is primarily determined by the seed. The seed tends to act like a slab reactor whose height is much greater than its thickness. Such a reactor is inherently subject to high axial flux peaking when controlled with absorbing type control rods moving as a bank, since the height of the rod bank for criticality is only a small part of the core height, and in the region containing the rods the flux falls away exponentially. The situation could have been improved by the use of burnable poisons which would have brought the rods farther out but, for reasons discussed below, burnable poisons were not used in the first seed. The means adopted to improve the power distribution was to program the rods in sequence as the core depletion proceeds. In this manner, most of the rods are either all in or all out, so that the axial flux distribution is more nearly that of a cosine. It was found that rod withdrawal patterns could be selected which also maintained satisfactory power distributions in the horizontal plane.

The use of burnable poison, as referred to above, was considered for the first seed as a means of reducing mechanical control and improving flux factors, but was rejected for several reasons. First and most important, the amount of excess reactivity margin available for fuel depletion was very uncertain, largely because of lack of experience with a core of this type. Second, there was little knowledge of the optimum burnup properties desired for the poison, since only extremely crude methods were available at that time to calculate the effects on reactivity of isotopic changes in the blanket. Finally, it was felt that the use of burnable poison would make the interpretation of depletion effects on the reactivity behavior of the core undesirably complex.

In parallel with the nuclear design studies, an extensive program of critical experimentation was carried out. This consisted of three phases:

(1) Determination of the nuclear parameters of lattices consisting of natural and slightly enriched uranium fuel elements in ordinary water.

(2) Slab experiments, using highly enriched seed sections bordered on each side by parallel slabs of blanket material.

(3) Full scale mockup of the complete core.

The nuclear parameters of the lattices were determined by a cooperative program of exponential experiments at the Brookhaven National Laboratory and of critical experiments on the TRX facility at the Bettis Plant. The fuel used in these experiments was uranium metal rods of three enrichments slightly above that of natural uranium, in order to reduce the amount of material required in the exponential experiments and to permit achievement of criticality in the Bettis work. The fuel material was exchanged between Brookhaven and Bettis. Close agreement was found between the results of the two series of experiments. The char-

acteristics of natural uranium lattices were determined by analyzing the slightly enriched lattices and extrapolating to the U^{235} content of natural uranium. To determine geometrical effects, the material was successively rolled down into fuel elements of three different diameters, and lattice measurements conducted in each case. In addition, Bettis conducted similar measurements on uranium oxide rods. As pointed out previously, the core reactivity is relatively insensitive to the blanket parameters. However, a knowledge of these parameters is necessary to permit calculation of the total power output of the blanket, the power distribution in the blanket, and the conversion ratio and consequent long-term reactivity behavior of the blanket.

In the early slab experiments at Bettis, the highly enriched fuel used was in the form of a uranium oxide powder contained in extruded plastic tape. This fuel form provided flexibility in changing the fuel loading, since the hydrogen density in the plastic was quite close to that of water. This fuel form proved quite useful also from the point of view of flexibility of fuel geometry. When the final mechanical design and seed fuel loading had been established, some seed fuel in the form of metal plates composed of uranium-zirconium alloy was procured to mock up the nuclear characteristics of this final design. This material was compared experimentally with the plastic fuel material used in the final nuclear mockup. The results of the plastic fuel experiments were confirmed by these experiments with the metal fuel.

The full core mockup made it possible to determine such important quantities as excess reactivity, adequacy of shutdown of the core by the control rods, power distributions, and blanket conversion ratio. One of the most important techniques developed in the critical program was a method for determining the excess reactivity of the core. The technique made use of the fact that core reactivity was determined primarily by the seed which, in this respect, was similar to a long, thin slab core. The shape of the seed suggested the probability that the horizontal and vertical bucklings of the core were separable, and that by varying these two bucklings simultaneously the core reactivity could be determined. The experiment consists of varying the vertical buckling by adjusting the moderator (water) height at which the core is critical with all control rods removed and, at the same time, varying the horizontal buckling by changing the amount of blanket material and, thus, the "reflector savings" of the seed. According to theory, if the bucklings are separable, the reciprocal cube root of the moderator worth (derivative of the reactivity of the core with respect to moderator height), when plotted against the core water height required for criticality, will give a straight line. The integration of the moderator worth curve obtained from this linear fit to the data gives the desired measure of the core excess reac-

tivity. Later developments of this technique showed that horizontal buckling can be changed by methods other than that of varying the amount of blanket material. One extensively used method was to provide a pattern of control rods which are completely inserted in the core, thus essentially breaking the core into a number of smaller cores with correspondingly larger horizontal buckling.

By these methods the excess reactivity of the first PWR core at room temperature was found to be approximately $17\frac{1}{2}$ percent, an amount which appears sufficient to permit operating the first seed for more than its design lifetime of 3000 full power hours. The calculated reactivity requirements for depletion, averaged over core lifetime, are two percent per thousand full power hours. The losses of reactivity measured on the operating PWR plant are approximately 2.6 percent for loss of reactivity in going from room to operating temperature, and 3.6 percent from the buildup of steady-state xenon and samarium poisoning, leaving more than 11 percent for depletion effects.

3–3. Detailed Physics Considerations

3–3.1 Power sharing between seed and blanket. The relative fraction of total core power output produced by the seed and the blanket, respectively, are of great practical interest from the standpoint of developing as much power as practicable from natural uranium.

The characteristics of the power sharing between the seed and blanket are illustrated by the following equation:

$$\frac{\text{Blanket power}}{\text{Seed power}} = \left(\frac{\epsilon k^B}{1 - k^B + \Delta k^B}\right)\left(\frac{k^S - 1}{k^S}\right), \tag{3–1}$$

where k^B is the infinite multiplication factor of the blanket, Δk^B is the loss of blanket multiplication factor due to blanket leakage, ϵ is the fast fission factor for the blanket, and k^S is the infinite multiplication factor of the seed.

The factor on the extreme right, which represents the fraction of neutrons produced in the seed that leaks into the blanket, is thus a measure of the efficiency of the seed as a neutron source. In the PWR, less than one-fourth of the neutrons originating in the seed leak into the blanket, the initial value of k^S being about 1.33. A higher value of k^S would, of course, materially increase the seed efficiency and, consequently, the blanket power. The limitation on the value of k^S is closely related to the maximum permissible power density, as can be seen from the approximate equations given below. Since k^S applies to a seed which is artificially poisoned by a uniform absorber in the seed to make the reactor just

critical, we may write

$$k^S \simeq 1 + M^2B^2, \tag{3–2}$$

where M^2 is the migration area in the seed and B^2 is an equivalent seed buckling. If we neglect the small axial leakage out of the seed,

$$B^2 \simeq \frac{1}{(t^S + R^S)^2}, \tag{3–3}$$

where t^S is the thickness of the seed annulus and R^S is the reflector savings (relatively constant) contributed by the blanket. From Eqs. (3–2) and (3–3) it is seen that k^S can be increased by reducing the seed annular thickness. This will increase the fraction of total core power produced by the blanket and reduce that in the seed. However, the seed volume is also reduced, and it is found that the decrease in seed power is insufficient to prevent a rise in power density in the seed. The power density permissible in the seed is restricted by the available coolant flow and temperature, and it is these factors which ultimately limit the value of k^S and, hence, of the blanket power fraction.

Although, as we have pointed out above, the k^S used in the power-sharing formula refers to a seed which is controlled by a uniform poison, it has been found that the power sharing between the seed and blanket is almost unaffected by the manner in which the seed is controlled. However, the power distribution in the blanket can be strongly affected. Thus, in studies in which the seed is controlled by a uniformly positioned bank of rods, the power in the seed is mostly concentrated in the lower portion of the core below the rod bank. On the other hand, the power in the blanket is much more uniform axially, the explanation being that fast neutrons originating in the lower seed region penetrate the thermal neutron absorbing rod region and cause multiplication in the upper blanket region. The total blanket power fraction is not materially changed from the case of a uniformly poisoned seed. Similarly, rod programming, while shifting the power distribution in the seed, does not significantly affect the total fraction of core power produced by the blanket.

The first factor on the right side of Eq. (3–1) represents the effect of the blanket multiplication factor on power sharing. The value of k^B for the uranium oxide blanket used in PWR is approximately 0.89 for operating conditions. Were it not for the term Δk^B, representing the effect of the blanket leakage, the blanket multiplication factor $1/(1 - 0.89)$ would be nearly 10 and would give a blanket power fraction of nearly 70 percent. However, the value of Δk^B is quite large. Calculations indicate its value to be approximately 0.10, of which about 0.07 is leakage to

the outside radial reflector, 0.02 is leakage to the outside axial reflector, and 0.01 is back leakage of thermal neutrons from the blanket to the seed. The latter term is explained as follows. Since the seed is more highly absorbing than the adjacent blanket regions, there tends to be a net thermal neutron current from the blanket to the seed. This effect, while relatively small from the standpoint of neutrons lost from the blanket, becomes important in designing very long-lived seeds which would have much higher fuel density and, hence, higher neutron absorption than the first PWR seed.

The presence in Eq. (3–1) of ϵ, the "fast effect," deserves some mention. This factor is a measure of the effect of direct fissioning in U^{238}. Its value for the PWR blanket is slightly over 1.04, meaning that the total fission neutron population is increased by more than 4 percent over that produced by thermal fissions. Taking into account the fact that the number of fast fissions must be sufficient to replace those neutrons absorbed to produce the fast fissions, it is found that nearly 10 percent of the total power in the blanket is produced by U^{238} fission.

3–3.2 Calculational techniques. The seed and blanket concept presents some unusual physics problems; solving them has led to an increased understanding of reactor physics principles and to the development of advanced calculational techniques. Study of a core of this type with multiplying regions of sharply different properties has brought about a more sophisticated and generalized treatment of the conventional models of core and reflector used in the treatment of criticality, reactor control, and reactor kinetics. Standard reactor design techniques proved inadequate for such problems as those presented by the large fast-neutron leakage from the seed into the blanket, the nonuniform resonance absorption and fast fission in the U^{238}, the large thermal neutron gradients, and the complex geometry of the core.

Even in the exploratory stage, the impracticability of computing the necessary multiregion problems by hand computers on a simplified one-dimensional, two-group basis necessitated the use of reactor simulator analogue computers and, as soon as available, of digital computers. As the design progressed, it became necessary to make much more detailed studies, requiring the use of such advanced computers as the NORC and the IBM-704. The needs of PWR physics helped to furnish the impetus to develop two-dimensional and transport-theory codes, which have been of great use to reactor physics work in general.

The need for at least a two-dimensional treatment of the seed and blanket design made it desirable, to save computer time, to try using a small number of groups to describe the slowing-down process. As mentioned above, two groups, one fast and one thermal, were used in the

exploratory studies. Later, the use of four groups was found more accurate, as would be expected in a core having regions of high buckling (in the seed) and high fast effect (in the blanket). For either the two-group or four-group systems the fast neutron diffusion coefficients are generated by a 54-group Fourier transform calculation in the P-1 approximation.

The cost and time involved in calculating all two-dimensional reactor problems in full detail is prohibitive, and for most cases is not possible with available facilities. Consequently, the problem of detailed power distribution has been broken down into two problems. The first involves the description of the reactor as having blanket material and seed material of uniform composition. In the second problem the seed and the blanket fuel assemblies are described separately, but in greater detail. The results of the detailed calculations are superimposed on the gross reactor calculation to give the complete description of the reactor. The validity of the superposition technique for determining the complete, detailed reactor properties has been established by a complete analysis of a typical example.

In the gross reactor calculation the materials associated with a 6-inch square region of the core have been assumed to be uniformly distributed; that is, the fuel, zirconium, and water are assumed to be uniformly distributed over the 6-inch modulus. The core then consists of a seed and blanket reactor in which both the seed and the blanket have uniform composition. From this type of problem the over-all power distribution between the seed and blanket and among the various seed and blanket regions is determined.

In the cell calculations used to describe the individual seed and blanket fuel assemblies, it has been assumed that the cluster in question is one of an infinite array of clusters, so that there is no net current of neutrons across the boundaries. The mechanical details of the cluster and surrounding water channels are described explicitly in this calculation.

A comprehensive description of the power distribution in the reactor explains the effects on the axial flux distribution of partially inserted control rods. Attempts to provide such three-dimensional descriptions of the reactor have been made in which separability of the neutron distributions in the horizontal and vertical directions is assumed. In addition, a crude three-dimensional diffusion theory code has recently become available. Results obtained thus far indicate that techniques of this sort can be developed to treat three-dimensional problems. However, for the nuclear design of the first PWR core it was necessary to proceed without such an analysis. Power distributions obtained from the critical mockup assembly, when the core was controlled by rod programming, have been used as a basis for estimating flux distributions in the PWR during operation.

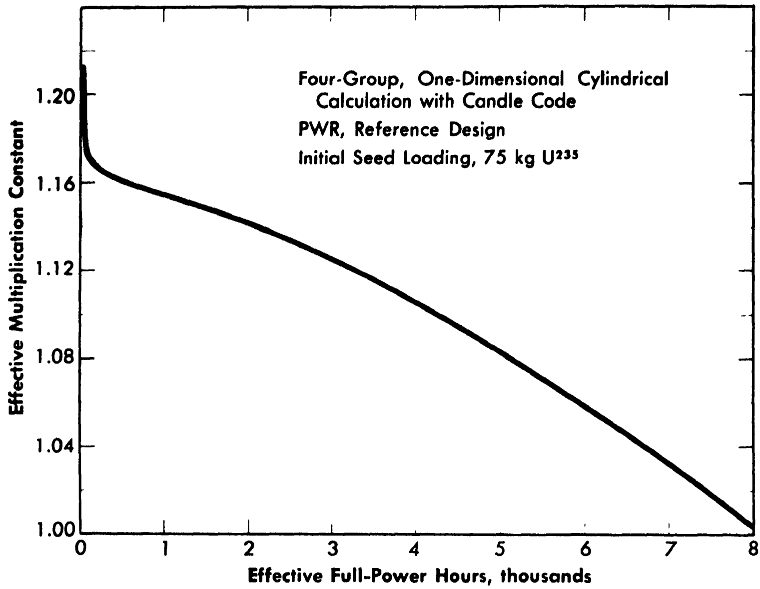

FIG. 3-1. Change in k_{eff} of PWR Core I as predicted by reactivity lifetime calculation.

3-3.3 Life history studies. In general, the lifetime calculations performed for the PWR were intended to answer several questions with regard to the design:

(1) What loading of U^{235} in the form of highly enriched uranium was required to provide the specified number of hours of full-power operation?

(2) How would the distribution of power between blanket and seed regions change as a function of core lifetime?

(3) What could be learned concerning intraregion shifts in power in the blanket as the blanket fuel properties change with time?

Only one-dimensional codes were available for the life history calculations. Also, it must be remembered that knowledge of the nuclear properties of the plutonium isotopes is still relatively meager. The results shown in Fig. 3-1 should, therefore, be considered as only approximate. The sharp initial drop in k_{eff} shown in the figure is caused by the buildup of the Xe^{135} fission product poison to an equilibrium value.

Corresponding studies of the variation of power distribution made it possible to select a coolant flow orificing system for the blanket which is expected to be adequate during the first seed lifetime.

3-3.4 Conversion ratio of blanket. In discussing the conversion ratio of the blanket it is very important to consider the influence of neutrons leaking from the seed. The following treatment of the conversion ratio of

the blanket at the start of life will illustrate this point, as well as the importance of "thermal" conversions—that is, thermal capture by U^{238} to form Pu^{239}.

From neutron balance considerations, it can be shown that

$$L^S = \eta_{25}\epsilon \frac{(1 - k^B + \Delta k^B)}{k^B}, \tag{3-4}$$

where L^S is the net number of neutrons leaking from the seed into the blanket per U^{235} nucleus destroyed *in the blanket* by either fission or parasitic capture, and η_{25} is the number of fission neutrons produced per U^{235} atom destroyed. The other symbols have been defined following Eq. (3-1).

From Eq. (3-4) we find that the seed supplies about 0.53 neutron to the blanket per each U^{235} nucleus destroyed in the blanket. Each such nucleus destroyed in the blanket also results in the creation of $\eta_{25}\epsilon = 2.19$ neutrons, making a total of 2.72 neutrons available in the blanket per U^{235} nucleus destroyed there.

The neutrons required in the blanket are as follows:

	Neutrons required
Replacement of each neutron absorbed by U^{235}	1.00
Compensation for leakage out of blanket per neutron absorbed by U^{235}	0.23
Compensation for absorption by water and structural material per neutron absorbed by U^{235}	0.33
Total needs	1.56

The difference between the total needs of 1.56 and the number available, 2.72, gives the average initial conversion ratio of 1.16 for the blanket. This number has been checked by measurements on the cold mockup critical.

It should be noted that of the total conversion ratio of 1.16, approximately 0.56 is due to thermal absorption by the U^{238} and the balance is due to resonance absorption. The value of 0.56 is equal to the initial ratio of the macroscopic cross section of the U^{238} to that of the U^{235}. In a slightly enriched reactor the ratio of U^{238} to U^{235} capture goes down inversely with the enrichment, reducing the thermal conversion ratio accordingly. A uniformly enriched core designed for long life would have a relatively high enrichment; consequently the initial thermal conversion ratio would tend to be low. On the other hand, increasing the loading of the seed to produce a long seed lifetime has practically no effect on the initial conversion ratio since, from Eq. (3-4), the input of neutrons from seed to blanket is not affected by seed characteristics.

A useful way of looking at the effect of core type on conversion ratio is to consider that there are essentially two ways of storing the extra fuel needed in a reactor to provide for lifetime depletion effects. The conventional way is to distribute the added fuel, in the form of higher enrichment, more or less uniformly through the core. This method requires compensating for the extra fuel by adding poison, usually in the form of control rods, to keep the reactor operating at critical. This added poison absorbs neutrons and produces a relatively small value for the conversion ratio.

A second method of storing fuel, the method used in the PWR, is to locate this fuel in a geometrical shape which has a large fast-neutron leakage, thus preventing the added fuel from producing a large amount of supercriticality. In this method of fuel storage the ratio of poison to fuel is essentially independent of the amount of highly enriched fuel added to the core, and consequently the value of the initial conversion ratio remains high.

3-4. Physics Results from PWR Operating Tests

Measurements have been made at Shippingport of reactivities and reactivity coefficients. The results of these measurements are presented briefly below.

3-4.1 Excess reactivity. Excess reactivity at 525°F was calculated to be 15.0 percent, and measurements yielded a result of 14.8 percent. At 70°F, the calculated and measured values were 17.6 and 17.4 percent, respectively.

3-4.2 Temperature coefficient of reactivity. At 525°F both the calculated and measured values of temperature coefficient of reactivity were -3×10^{-4} per °F. The room-temperature value was measured as -0.5×10^{-4} per °F and calculated to be about -0.8×10^{-4} per °F.

3-4.3 Pressure coefficient of reactivity. The calculated value of pressure coefficient of reactivity was $+2.2 \times 10^{-6}$ per psi. The average value of the measurements was $2.4 \pm 0.6 \times 10^{-6}$ per psi.

3-4.4 Other results. The reactor proved able to start up with peak xenon built up after shutdown from an extended full-power run. The loss of reactivity at the start of life from hot clean critical to equilibrium xenon was measured as approximately 3 to 4 percent and from equilibrium to maximum xenon as 6 percent.

No evidence of steady-state power asymmetry or xenon oscillation has been found. There had been some concern that such phenomena might occur as a result of the decoupling of various parts of the seed from each other. While such asymmetry can be produced by control rod misalignment, the effect is small enough that it does not constitute a problem in the normal operation of the core. In addition, it appears that any flux asymmetry which may be established during transient operation can be corrected by means of appropriate control rod manipulation.

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CHAPTER 4

REACTOR

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REACTOR*

4-1. Introduction

This chapter presents a design description of the Shippingport reactor. It includes: (1) description and illustrations of the reactor components, the fuel-handling equipment, and service facilities; (2) discussion of the reactor core thermal design considerations and its capabilities; (3) tabulations of reactor characteristics and design parameters.

A number of factors, introduced in Chapter 1, strongly influenced the design of the reactor. Two have been selected for detailed discussion to illustrate their influence on design: (1) reliability and repairability, and (2) seed-blanket core arrangement.

4-1.1 Reliability and repairability. Most of the equipment in the Shippingport reactor is developmental. Accordingly, several steps were taken during its design and development to ensure maximum reliability and to minimize the effects of failure on plant operation and availability. These included: (1) conducting extensive proof testing of reactor design features; (2) incorporating in the design a number of backup protective features.

The proof tests were instrumental in assuring reliability of the reactor. For example, while testing a seed fuel assembly, the stainless steel latch used to hold the assembly in the core cage (described later in this chapter) was found unsatisfactory. A full-scale prototype seed fuel assembly (with its accessories, including the control rod, shroud, shafting, and control rod drive mechanism) and a full-scale prototype blanket fuel assembly were tested for seven weeks in a hot loop at design flow rate and temperature. While this test assembly was heating up to operating temperature, differential expansion occurred between the stainless steel latch and the Zircaloy seed assembly, slightly loosening the two parts. When subjected to rated design flow, the latch assembly caused severe fretting of the adjacent Zircaloy. Accordingly, the design of the seed assembly was modified with a spring to compensate for differential expansion and prevent movement between the two pieces during operation.

Backup protective features were provided in the detailed design of the reactor to mitigate the consequences of failures. For example: (1) All

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bolts and mechanical fasteners inside the reactor were locked in so that in case of a failure, they would be held in place. One reason for this was to prevent parts such as bolt heads from interfering with the operation of other components such as pumps. (2) Backup seals consisting of metallic compression rings were provided on the stainless steel housings of the pressure vessel head to prevent leakage in case the welded leakage barrier failed. The stainless steel housings for the pressure vessel head extend downward to the interior of the reactor vessel head where they are welded to the cladding, thereby forming the primary leakage barrier (see Fig. 4–3). (3) Backup protection was provided to prevent wetting of electrical components on top of the reactor vessel head if waterproofing failed. All equipment on the head is made waterproof so that the reactor head area can be flooded during refueling. Electrical components are pressurized with air to ensure against leakage should the leakage barriers be faulty.

In addition to these steps, provisions were made, wherever possible, for performing corrective and preventive maintenance on reactor components. As pointed out in Chapter 1, the design of the reactor, fuel-handling equipment, and service facilities is such that any fuel assembly can be readily removed, relocated, or replaced through ports provided in the reactor vessel head. The design of specific components was also influenced by maintenance considerations. For example: (1) All control drive mechanism electrical parts are located for easy access outside the pressure boundary of the reactor vessel. (2) All moving parts including control rod drive mechanism parts, control rods, and their accessories such as shrouds and scram shafts can be easily removed by opening ports in the pressure vessel head. (3) The majority of the core instrumentation can be removed easily through the ports in the pressure vessel head.

4–1.2 Seed-blanket core arrangement. The seed-and-blanket core uses a relatively small amount of enriched uranium and a large amount of natural uranium. As shown in Fig. 4–1, the enriched uranium (approximately 90% U^{235}) is contained in 32 assemblies collectively called the "seed," while the natural uranium is contained in 113 assemblies collectively called the "blanket." Seed assemblies are arranged in a square annular array with blanket assemblies both inside and outside. Thus a core is produced in which neutrons leaking from the highly enriched seed cause fission in the natural uranium of the blanket assemblies. The seed fuel contains approximately 165 lb of U^{235}. The blanket is made up of 14.2 tons of natural uranium in the form of uranium dioxide (UO_2). The U^{235} in the blanket, being dispersed and associated with the matrix U^{238}, cannot by itself sustain a chain reaction. Furthermore, the seed assemblies in their square annular array, even in the presence of water and without their control rods, cannot by themselves sustain a chain reaction. The

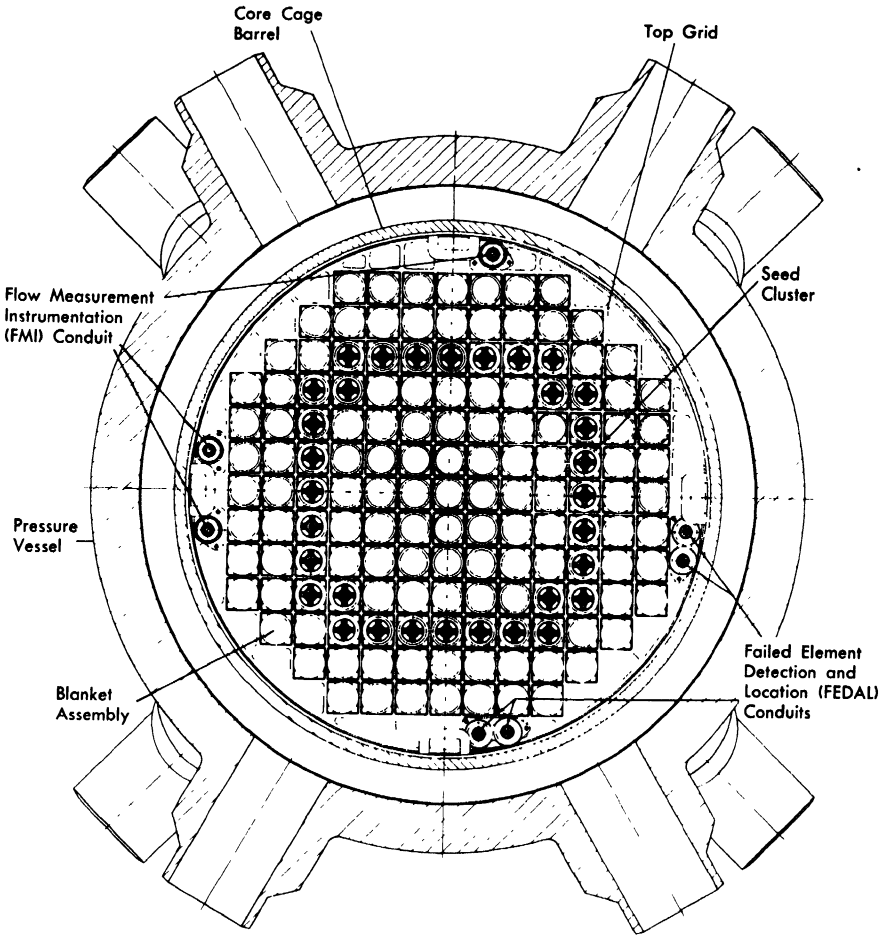


FIG. 4-1. PWR, cross section through outlet nozzle.

neutron reflecting characteristics of at least a portion of blanket material is required to sustain the chain reaction.

Many variations of the seeded reactor are possible, and the seed geometry chosen was based on extensive studies showing this to be favored from the standpoint of nuclear factors and ratio of blanket to seed power generation. The choice of this type of core allowed considerable freedom in permitting changes to be made in the course of its design. The seed could be repositioned on the basis of the latest analytical and experimental data; this was actually done throughout the design.

The seed-blanket arrangement permits a relatively simple reactor structural design for the amount of flexibility and accessibility provided. The smaller number of control rods required and the annular configuration of

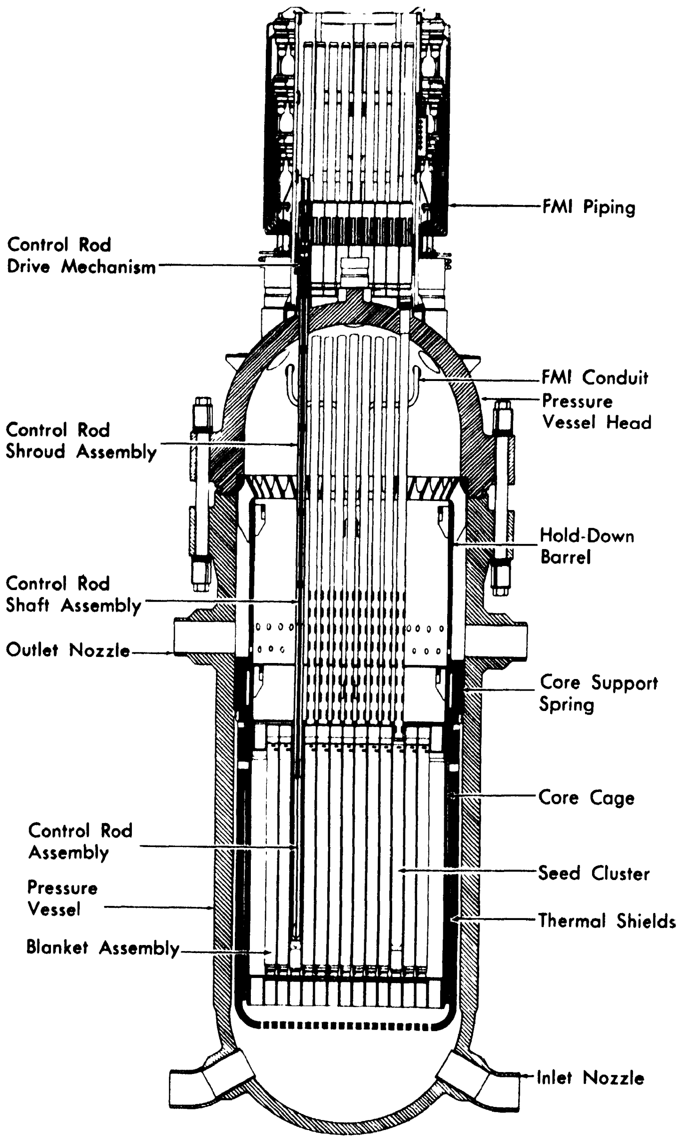


FIG. 4-2. PWR, centerlines section.

the seed and its associated control rods permit (1) sufficient space within the reactor vessel to remove, replace, or rearrange the fuel assemblies, and (2) a head penetration pattern with 46 openings that meets code requirements and provides a head equipment arrangement such that there is ready access through the refueling ports.

Consistent with the objectives stated in Chapter 1, to provide as much experience as possible with this type of core the reactor contains a relatively large amount of core instrumentation. This instrumentation permits measuring selected core parameters while the reactor is in operation and checking results against calculated values. Quantities measured include fuel-element temperatures, flow through individual assemblies, power distribution, and detection and location of failed elements.

Figure 4–2 shows a longitudinal section of the Shippingport reactor. Its operating characteristics are given in the Table "Shippingport Plant and Reactor Characteristics" in the Appendix. Details of the reactor vessel and core components, the thermal capabilities of the core, and the fuel-handling and refueling equipment are discussed in the following sections of this chapter.

4–2. Description of Vessel and Core Components

The major components of the reactor assembly, shown in Fig. 4–2, are: the reactor vessel and closure head; the thermal shields; the core hold-down barrel; the core assembly, including fuel components and core cage; and the control rods, their drive mechanisms, and accessories. These components are described in detail below.

4–2.1 Reactor vessel. The reactor pressure vessel, shown in Fig. 4–3, is essentially cylindrical, with an inside height of 375 in., an internal diameter of 109 in., and a nominal wall thickness of $8\frac{3}{8}$ in. The vessel has a bottom hemispherical head and a flanged hemispherical top head. Four inlet nozzles in the bottom head receive water from the reactor coolant loops, and four outlet nozzles near the middle of the vessel discharge the heated water back into the coolant loops. The top, or closure, head is removable for core replacement. It also provides access through ten fuel ports to replace any individual fuel assemblies without removing the closure head. The reactor vessel, besides housing and supporting the reactor core, also supports the instrumentation system and control rod drive mechanisms.

The vessel is formed of manganese-molybdenum carbon steel plates and forgings (ASTM-SA-302, Grade B) with a $\frac{1}{4}$-in. stainless steel (AISI type-304L) cladding. The 302 material was selected because it was the highest strength carbon steel allowed by code. The weight of the complete

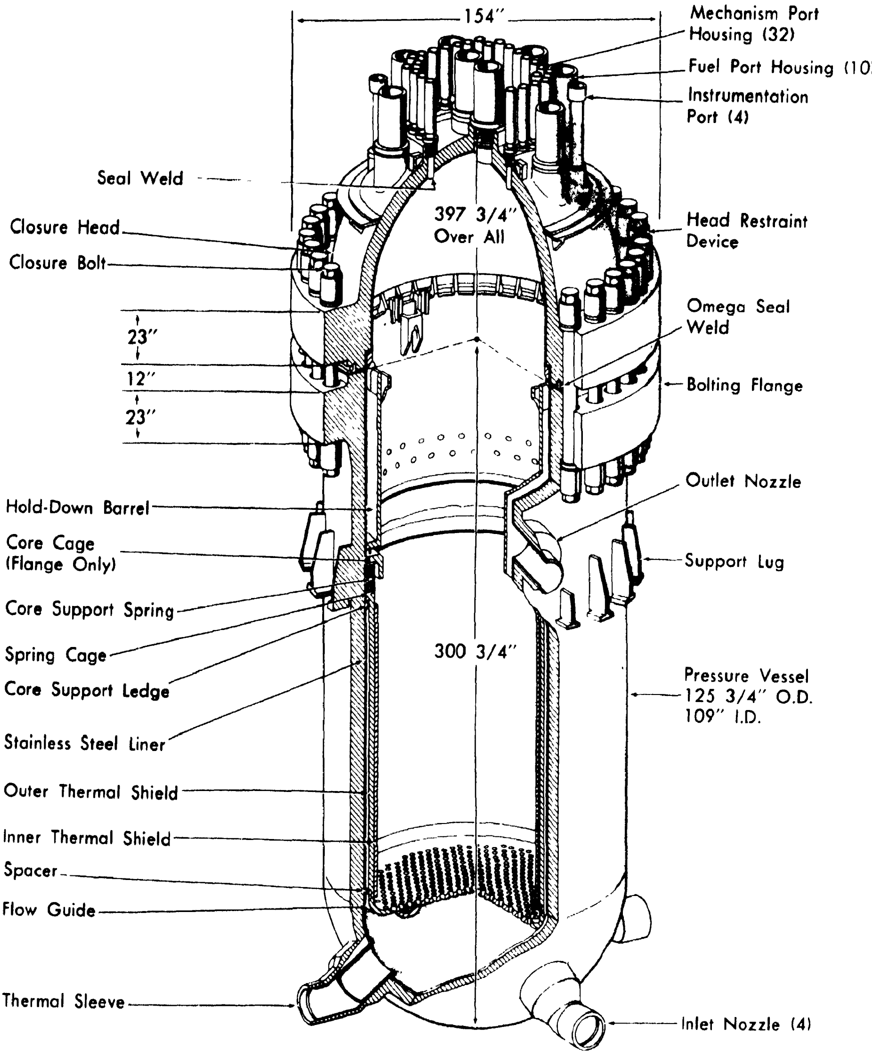


FIG. 4-3. Reactor pressure vessel, cutaway view.

vessel, including the closure head and thermal shields, is approximately 264 tons.

The vessel shell and the bottom hemispherical head were made from plates to which the stainless steel cladding had been roll bonded. These were the thickest clad plates ever roll bonded in the United States. The top head, however, because of its 10-in. thickness, could not be roll bonded with existing equipment. Its cladding, instead, was deposited by machine welding.

No code exists that fully covers the design and fabrication of this pressure vessel. However, it meets the requirements of Section I, Power Boilers, of the ASME Boiler and Pressure Vessel Code and also the Pennsylvania state code. In the regions not covered by the code, extensive analytical and experimental studies were made to ensure that the vessel was reliable and operationally safe. A full-diameter, reduced height pressure vessel and full-size closure head were tested to demonstrate the adequacy of the final design and to establish acceptable heating and cooling rates.

The vessel, as designed, is adequate for the following conditions:

(1) An internal design pressure of 2500 psi.
(2) An operating (nominal) pressure of 2000 psi.
(3) A hydrostatic test pressure of 3750 psi.
(4) A design temperature of 600°F.
(5) An allowable heat-up rate of 70°F/hr.
(6) An allowable cool-down rate of 50°F/hr.
(7) Short, rapid temperature changes as follows, with appropriate waiting periods to be consistent with the above:

(a) An increase of 50°F in 45 sec.
(b) A decrease of 30°F in 45 sec.

Vessel shell and attachments. The shell is a right circular cylinder formed of three cylindrical courses joined by circumferential welds. Each course, in turn, is made of two semicircular plates joined by longitudinal welds. The longitudinal welds in the three courses were oriented 90 degrees from each other to avoid possible crack propagation.

A bolting flange, 23 in. high and 154 in. in outside diameter, is attached to the upper end of the vessel shell. It is drilled with 42 equally spaced holes to receive bolts that attach the closure head to the vessel.

The bottom head, attached to the vessel shell, is hemispherical, with a wall thickness of 6.2 in., including 0.2 in. of cladding. The head was made of four pie sections and a central dome-shaped piece formed and welded together with locally deposited cladding over the weld area. This head is designed in accordance with the ASME Power Boiler Code, Section I, with an additional thickness to allow for piping loads. It is penetrated by four inlet nozzles spaced 90 degrees apart. The centerlines of these radial nozzles are perpendicular to the hemispherical head.

The inlet nozzles are lined with stainless steel thermal sleeves where they enter the interior of the vessel. The primary purpose of these sleeves is to prevent thermal shocks induced by cold water coming into contact with the nozzles, which might occur when an idle loop at lower than operating temperature is put into service. The outlet nozzles are located

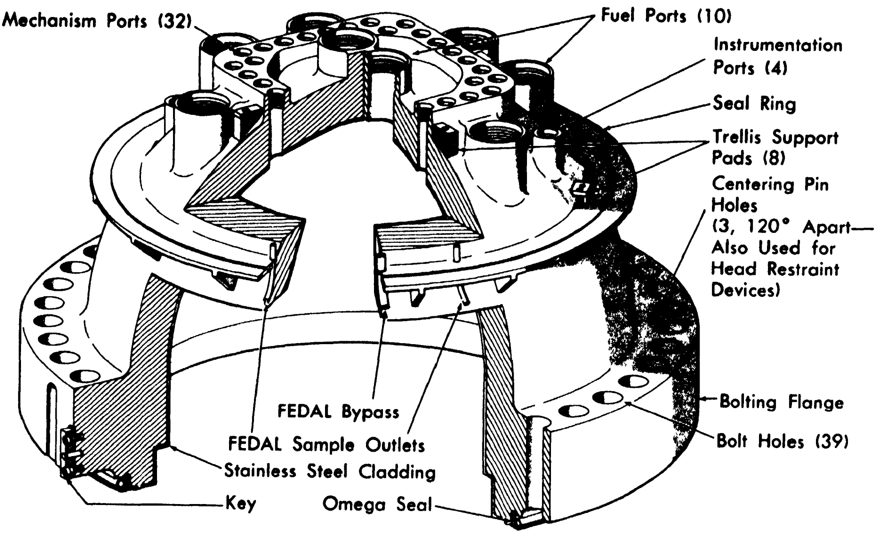


Fig. 4–4. Closure head.

on the cylindrical portion of the vessel $66\frac{15}{16}$ in. from the joint between closure head and vessel. In the design of both the outlet and inlet nozzles, the area removed for the vessel penetrations is added to the cross-sectional area of nozzles in accordance with ASME code requirements. All the nozzles are internally clad with stainless steel.

A support ledge is located approximately 34 in. below the centerlines of the outlet nozzles. This ledge is built up from low alloy steel weld metal covered with a stainless steel weld metal buildup to provide a continuous noncorrosive surface. This ledge supports the thermal shields which, in turn, support the core support springs, core, and hold-down barrel.

The vessel is supported 3 ft below the centerlines of the outlet nozzles by 24 support lugs, each $8\frac{1}{2}$ in. wide and 15 in. high, equally spaced and welded to the outside surface of the vessel. The lugs rest on horizontal radial pins which allow the vessel to expand and contract without producing high stresses in the vessel and supporting structures.

Closure head. The closure head (shown in Fig. 4–4), when attached to the shell, completes the reactor vessel proper. The closure head, like the bottom head, is designed to satisfy the ASME Boiler and Pressure Vessel Code, Section I, with additional thickness added for penetrations. There are a total of 46 reinforced penetrations: 32 of 4-in. diameter in a square pattern to accept and support the control rod drive mechanisms; ten of 10-inch diameter for refueling ports; and four of 4-in. diameter to accommodate instrumentation leads. These dimensions include a $\frac{1}{4}$-in. stainless steel cladding. Because of the number and sizes of penetrations, photo-

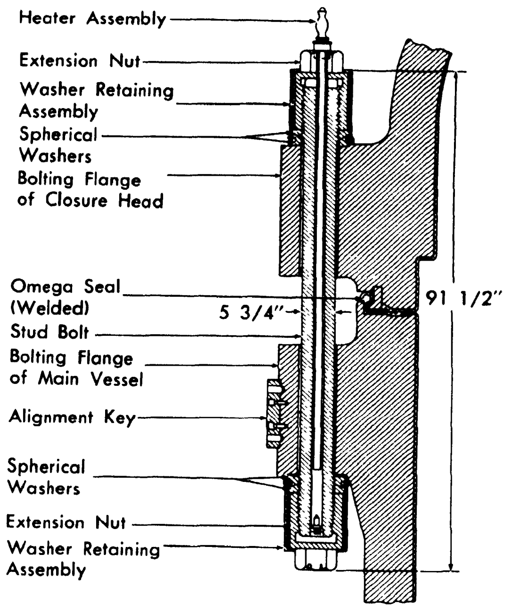


FIG. 4-5. Seal and closure assembly.

elastic studies were required to establish the feasibility of the penetration pattern. Tests with the full-size test head confirmed the adequacy of the final design.

The head is dome shaped, with a $10\frac{1}{4}$-in. wall thickness, including $\frac{1}{4}$ in. of stainless steel. This head was formed in a manner similar to that described for the bottom hemispherical head. Reinforcement around each one of the penetrations was built up from deposited weld metal.

To each of the penetrations on the head are added stainless steel housings fastened to the head by threaded connections, as shown in Fig. 4-4. Reduced sections of the housings extend downward through the penetrations and are welded to the head cladding to complete the cladding and provide a leaktight barrier. A bolting flange, similar to that of the vessel shell, surrounds the lower part of the closure head.

As Fig. 4-4 shows, a ring of $1\frac{3}{4}$-in. thick steel plate is welded to the outer surface of the head. During replacement of fuel assemblies, a mating gasketed seal ring is lowered over this surface, spanning the area between the seal ring on the closure head and the upper flange of the reactor container, as shown in Fig. 4-25. Ninety tapped holes in the seal ring on the closure head accommodate bolts securing the gasketed seal in place, making a water seal between the space below the reactor closure head and the space above it. This arrangement allows flooding of the space above the seal ring with water without letting water enter the space below

the seal ring. With this facility, fuel assemblies can be replaced under water through ports in the head.

Seal and closure assembly. The seal and closure assembly, Fig. 4–5, forms a pressure-tight, leaktight seal between the closure head and the top of the vessel shell. During operation, the seal weld prevents leakage but is not intended to make the joint strong enough to withstand the end force on the head. This strength is provided by the closure bolting.

As shown in Fig. 4–5, one side of an omega toroidal steel ring is welded to the under surface of the closure head adjacent to the point of contact with the top of the vessel shell. After the closure head has been placed in position, but before the closure studs are installed, the other side of the ring is welded to the top edge of the shell. It thus forms a watertight seal that is able to accommodate flange deflections and stud relaxations far beyond those allowed in a gasketed joint.

The seal ring is not disturbed when fuel assemblies are replaced through penetrations in the closure head. It is cut only when it is necessary to remove the closure head from the vessel. The ring is so constructed that it can be reused after having been cut for removal of the head. A special cutting machine has been built both to cut the seal and to prepare it for reuse.

Mechanical strength for the closure is provided by 39 stud bolts and three restraint bolts that pass through the bolting flange of the head and vessel shell. The bolts are made of alloy steel type SA-193 (Modified). The stud bolts are retained by cap nuts seated on mated spherical washers at each end. These washers prevent a bending stress from being exerted on the stud bolts by flange rotation. Each stud bolt is 6 in. in diameter and, when the cap nuts are installed, the assembly is $7\frac{1}{2}$ ft long and weighs approximately 820 lb.

Three restraint bolts (Fig. 4–6), placed 120 degrees apart, prevent any lateral movement of the head during operation. Tapered sections near the center of each bolt fit into segmented, tapered collars that, in turn, fit into the bolt holes in both flanges. When the restraint bolts are tightened, the tapered sections force the collars to expand and fit the bolt holes tightly, holding the closure head securely centered over the vessel shell.

The stud bolts are drilled through their centers to accommodate heating elements that, during installation, are used to expand the bolts before the nuts are tightened. As a result, the bolts are prestressed when the unit is at ambient temperature, and a tight joint between flanges is thus obtained at operating temperature. Specifically, the total prestress in the stud bolts is greater than the total end thrust on the head resulting from internal pressure. This prevents them from lengthening under pressure, which would leave the head free to move, causing excessive flexing of the seal and misalignment of the control rods between the head and the core.

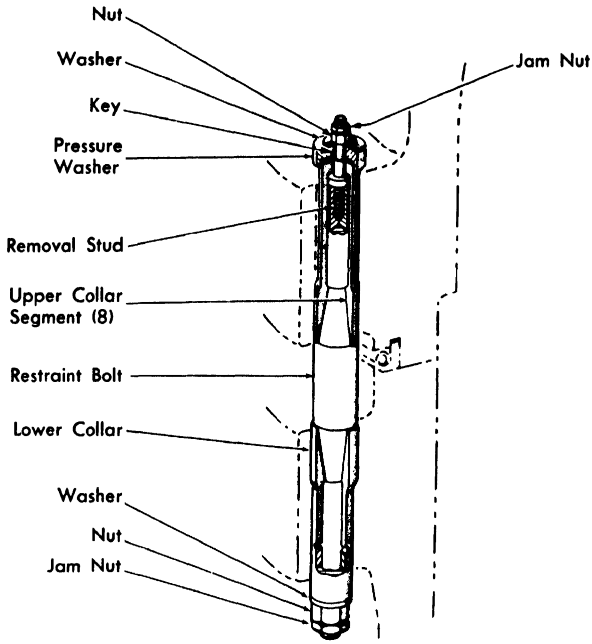


FIG. 4-6. Restraint bolt.

During heat-up and cool-down, the temperature of the bolts, and hence their expansion and contraction, lags somewhat behind that of the vessel and closure head, limiting the allowable heating and cooling rates of the pressure vessel. The prestress is of such magnitude that stresses in the bolts during heat-up will not be excessive, and during cool-down the head will not be lifted from its seat.

A groove in the bearing face of the head allows pressure equilization between the inside of the vessel and the inside of the omega seal, permitting the coolant inside the seal to expand when the system is brought up to operating temperature.

4-2.2 Thermal shields and flow guide. Thermal shields surrounding the core reduce radiation and therefore minimize heat generation in the pressure wall. The thermal shields consist of two stainless steel cylinders, one inside the other, supported from the core support ledge inside the vessel shell. The inner cylinder is 1 in. thick, the outer cylinder 3 in. thick. The shields create an annular space of $1\frac{1}{2}$ in. between the outer surface of the outer cylinder and the vessel wall, an annular space of $\frac{3}{4}$ in. between the cylinders, and a similar annular space between the inner cylinder and the core assembly. The annuli made up by the shields, vessel wall, and core

assembly provide flow channels through which coolant passes to cool these structures.

A flow guide or baffle, penetrated by 241 holes of 1.9-in. diameter, is attached to the outer thermal shield cylinder (see Fig. 4–3). This flow baffle distributes flow evenly to the core and also provides the potential for promoting flow through the thermal shield cooling annuli.

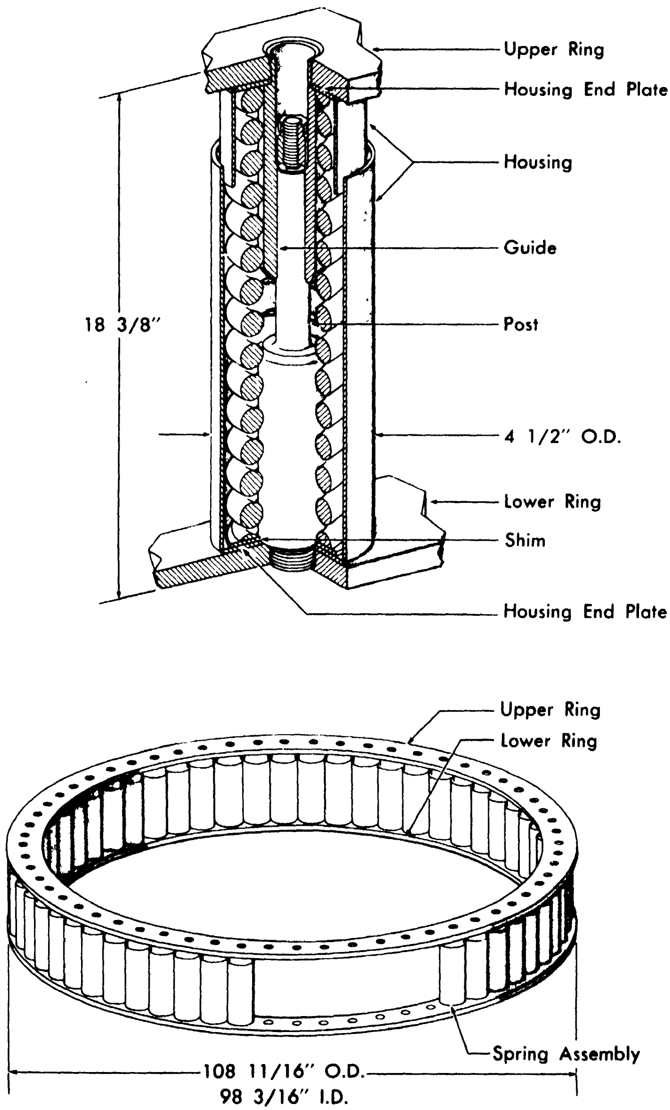


FIG. 4–7. Core support spring assembly.

4-2.3 Core support springs. The core assembly is supported by, and clamped against, the hold-down barrel by an assembly of 60 helical springs. Springs are used rather than a rigid support to allow both for machining tolerances and for differences in thermal expansion between various members of the reactor assembly. Figure 4-3 shows the arrangement within the pressure vessel. A flange at the top of the core assembly rests on top of the spring assembly. The bottom of the spring assembly is supported by the flange at the top of the thermal shielding, which in turn rests on the core support ledge on the inner surface of the pressure vessel.

The springs are made of Inconel X; all other parts are made of type-304 stainless steel. As Figs. 4-3 and 4-7 show, the springs are mounted vertically between upper and lower rings. The springs, spaced six degrees apart, are centered around a guide and guidepost connecting the upper and lower rings. The upper end of the post fits into the hollow center of the spring guide so that the two units telescope under pressure, allowing a total compression of $2\frac{5}{8}$ in. This arrangement vertically aligns the upper and lower rings and the springs. Each spring is enclosed in a two-part telescoping housing to prevent fragments from entering the coolant stream should it fail.

4-2.4 Core hold-down barrel. The core assembly is clamped against its support springs by the core hold-down barrel, a hollow cylinder of stainless steel. Figure 4-8 shows a flange at the upper end of the barrel which bears against a shoulder (Figs. 4-4 and 4-5) on the lower part of the closure head. The lower end of the barrel bears against the upper flange of the core assembly. During operation, the hydraulic thrust acting on the reactor core is transmitted through the hold-down barrel to the closure head.

The hold-down barrel, including flanges, is approximately 7 ft 4 in. high and 9 ft $\frac{1}{2}$ in. in outside diameter. The barrel proper is a cylinder with an outside diameter of 8 ft $\frac{1}{2}$ in. and a wall thickness of $1\frac{7}{8}$ in. The flange at the lower end of the barrel is an integral part of the cylinder; the upper flange is not an integral piece but is offset from it and supported by gusset plates welded to the upper and outer surfaces of the cylinder. This arrangement allows the upper flange to transmit the thrust load to the closure head of the vessel, but permits the coolant water to flow outward in the space between the gusset plates. The hold-down barrel is pierced by 32 radial holes, $1\frac{3}{8}$ in. in diameter, at the level of the centerline of the outlet piping, and another 32 holes, $1\frac{3}{4}$ in. in diameter, 12 in. above this centerline. Each hole is directed at one of the control rod shrouds so that, in case of loss of coolant, emergency coolant can be injected into the reactor outlet nozzles and through the holes to ensure cooling of the entire core. In case of loss of coolant flow, natural convective

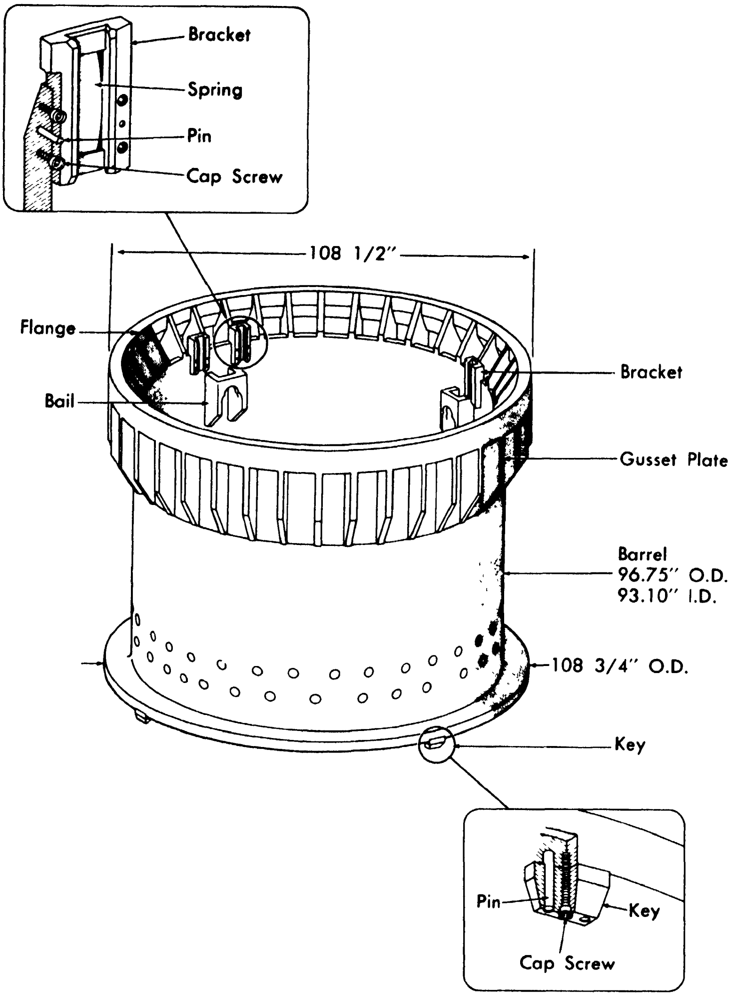


FIG. 4-8. Core hold-down barrel.

flow will proceed from the core, through these holes, and into the outlet nozzles. These holes will provide adequate cooling of the core if loss of coolant or loss of coolant flow occurs.

4-2.5 Flow pattern. Figure 4-9 shows the flow path that the coolant takes within the pressure vessel. Flow paths among individual fuel assemblies have been omitted for simplicity.

Coolant water enters the lower plenum chamber through the four inlet nozzles at the bottom of the reactor vessel. Approximately 85% of the total flow passes through the flow baffle and is properly distributed to the

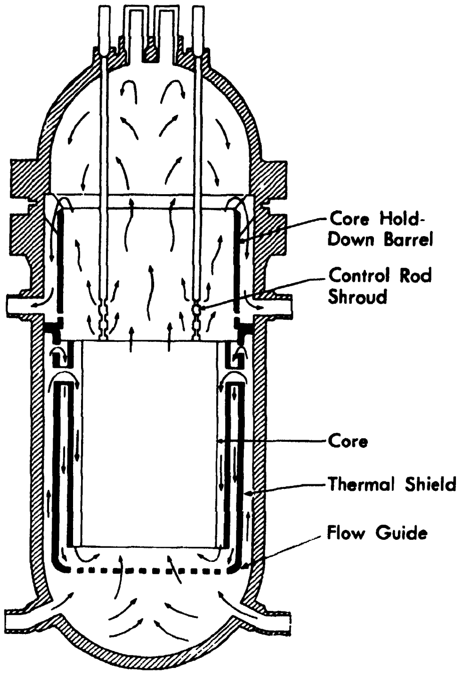


FIG. 4-9. Flow pattern in pressure vessel and thermal shields.

core. The pressure differential across the flow baffle provides the potential to direct the flow of the remaining 15% of the water through the cooling annuli. This 15% is directed through the annulus between the reactor vessel wall and outer thermal shield and is then equally divided between the two inner annuli, where it is directed downward, joining the main body of flow at the entrance to the core.

Exit water from the blanket fuel assemblies discharges directly into the upper plenum chamber. Exit water from the seed first enters the shrouds that enclose the control rods when they are pulled out of the core, then flows outward through holes in the lower parts of the shrouds, joining the blanket flow in the upper plenum. The main portion of the flow is then directed upward by the hold-down barrel, outward through spaces between the gusset plates at the upper end, and finally downward on the outside of the hold-down barrel to the outlet nozzles, where it again enters the main loops. A small portion of the flow passes through the two rings of 32 holes located near the outlet ducts.

4-2.6 Thermal insulation and canning. The vessel and closure head (Figs. 4-10 and 4-11) are thermally insulated by Fiberglas packed to a density of $4\frac{1}{2}$ lb/ft^3 and enclosed in cans to prevent the material from

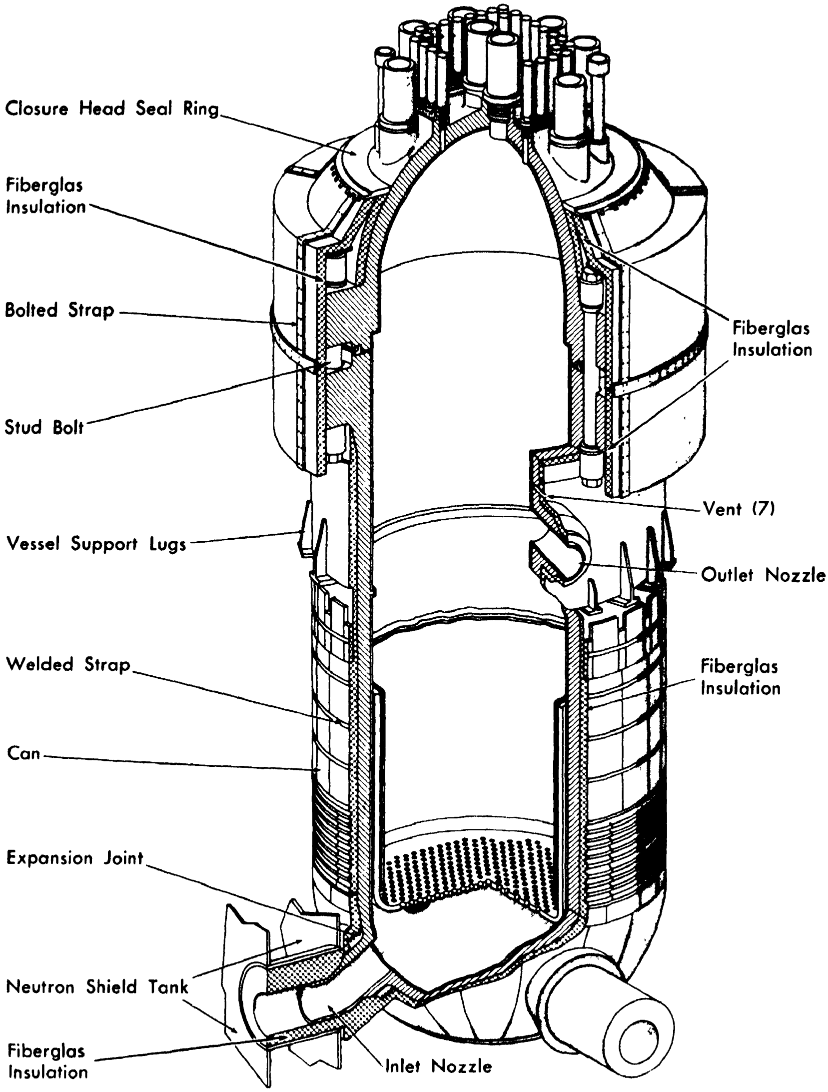


FIG. 4-10. Vessel shell insulation and canning.

coming in contact with water during refueling operations. The insulation is made of Fiberglas blankets having hexagonal wire mesh facing on the side against the shell and expanded metal on the other side. The free thickness of the blankets is $4\frac{1}{2}$ in., which is compressed to 4 in. when canned; near the closure bolts, $2\frac{1}{2}$-in. blankets are used, compressed to 2 in. when canned.

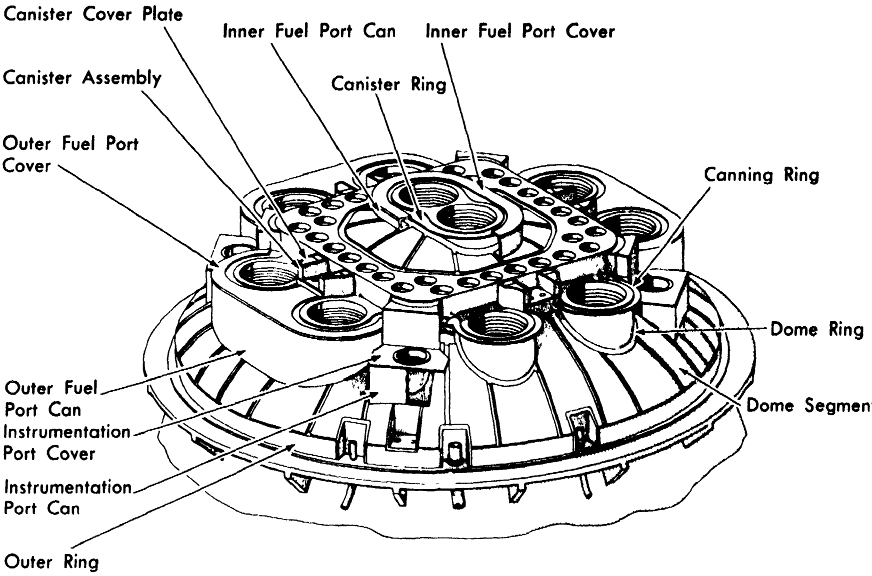


FIG. 4–11. Head insulation and canning.

The canning material is 20-gage (U. S. Standard) stainless steel throughout. Once placed in position, the insulation on the lower part of the vessel need not be disturbed, and the canning is therefore permanently welded together. In the region of the closure head, however, the canned insulation is removable in sections to facilitate operations on the head. In regions of the head where the form is so irregular as to make canning impractical, bare Fiberglas blankets are used.

4–2.7 Tabulated data. Data on the reactor vessel are presented in Table 4–1.

TABLE 4–1

REACTOR VESSEL DATA*

Complete vessel	
Over-all height, including stainless steel head penetration housings	$417\frac{1}{1}\frac{5}{6}$
Diameter of bolting flanges	154
Weight, including closure head and thermal shields	264

* All dimensions are in inches and all weights are in tons.

TABLE 4-1 (*continued*)

Vessel shell	
Over-all height including bottom head	$300\frac{3}{4}$
Inside diameter	109
Outside diameter	$125\frac{3}{4}$
Bottom head inside radius	$55\frac{9}{16}$
Bottom head outside radius	$61\frac{3}{4}$
Nozzle inside diameter	15
Nozzle outside diameter	18
Weight	150
Closure head	
Over-all height, including stainless steel head penetration housings	$117\frac{3}{16}$
Inside radius	$51\frac{1}{2}$
Outside radius	$59\frac{3}{4}$
Weight	85
Thermal shields	
Outer thermal shield	
Over-all height, including flow baffle	$150\frac{1}{2}$
Flange diameter	$108\frac{3}{4}$
Cylinder outside diameter	106
Cylinder inside diameter	100
Inner thermal shield	
Over-all height	141
Outside diameter	$98\frac{1}{2}$
Inside diameter	$96\frac{1}{2}$
Weight of complete assembly, including flow baffle	29

4–2.8 Core assembly. The core (Fig. 4–12) is the active portion of the reactor, containing the fuel. Its main components are the core cage, which houses and supports all the other components; the fuel assemblies; the sensing elements of the flow rate and temperature instrumentation; connections for the failed element detection and location system; and the control rods.

The fuel assemblies, as shown in Fig. 4–12, are supported within the core cage by a bottom support plate and a top grid and are so attached that they may be removed individually. This allows handling of individual damaged or spent fuel assemblies without disturbing the remaining assemblies. It also allows the rearrangement of assemblies to equalize burnup of the fuel in order to maintain even power distribution in the core.

Fuel assemblies arrangement. The highly enriched, or seed, region (approximately 90% U^{235}) contains 32 assemblies arranged in a hollow square, with the assembly in each corner set inward from the others. Control rods are placed only in the seed, since it is only in these assemblies that the

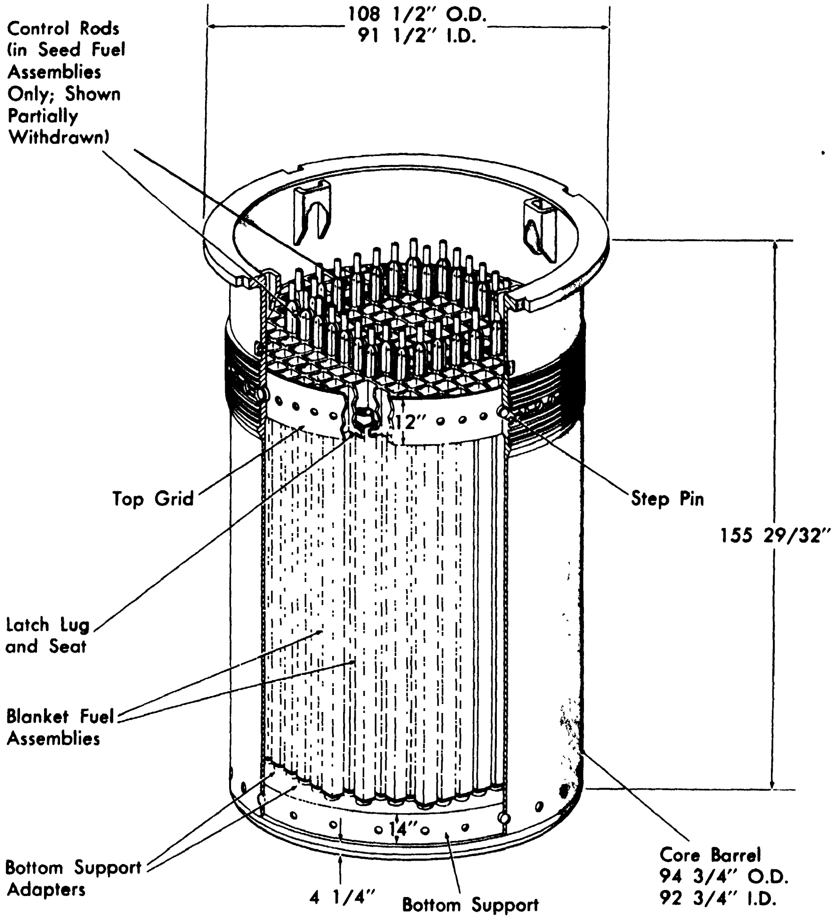


FIG. 4-12. Core assembly.

concentration of U^{235} is high enough to exceed the fission rate necessary to sustain the chain reaction.

The natural uranium, or blanket, assemblies occupy the areas inside and outside the seed square. There are 113 blanket assemblies, 45 inside the seed square and the rest surrounding the seed. The blanket assemblies, as shown in Fig. 4-13, are divided into four regions according to the radial thermal neutron flux variation. Different flow requirements thus exist for each region. Since the seed assemblies dictate the pressure differential across the core, restrictions to the flow are placed in the blanket assemblies to obtain the desired flow distribution among regions. These restrictions to flow consist of multihole orifices. Within any region, including the seed, the flow is divided equally among all the assemblies.

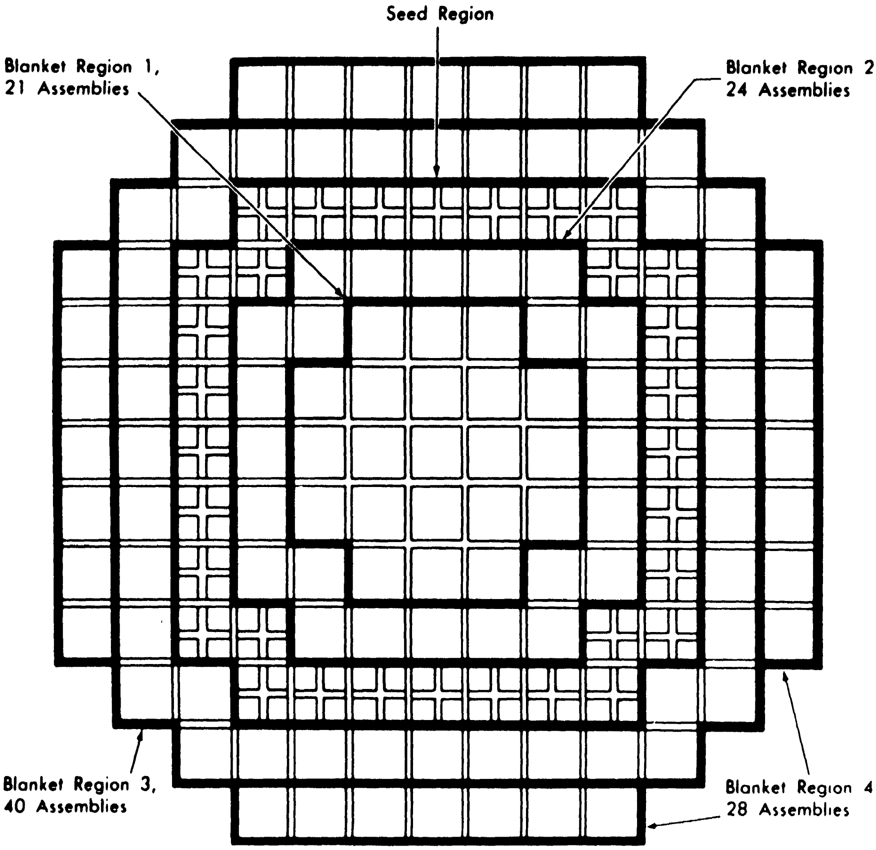


FIG. 4–13. Blanket assembly regions.

A metallic alloy of highly enriched U^{235} in the form of plates was chosen for the seed because of its previous good performance in pressurized water reactors. The fuel in the blanket region is natural uranium in the form of UO_2 pellets clad in Zircaloy tubes. As discussed in detail in Chapter 5, this fuel was chosen because it is chemically inert to hot water, is easily fabricated, and is satisfactory from a radiation damage viewpoint for a relatively high burnup. All seed assemblies are 5.5 in. square in cross section, and all blanket assemblies are 5.6 in. square; all assemblies are on 6-in. centers.

Blanket assembly. The blanket assembly, shown in Fig. 4–14, contains and supports the blanket fuel, consisting of natural uranium dioxide. The uranium dioxide is enclosed in tubes, assembled parallel to each other in groups called bundles. Seven such bundles make up an assembly. The stacked height of the bundles is $71\frac{3}{4}$ in. The active height of the uranium

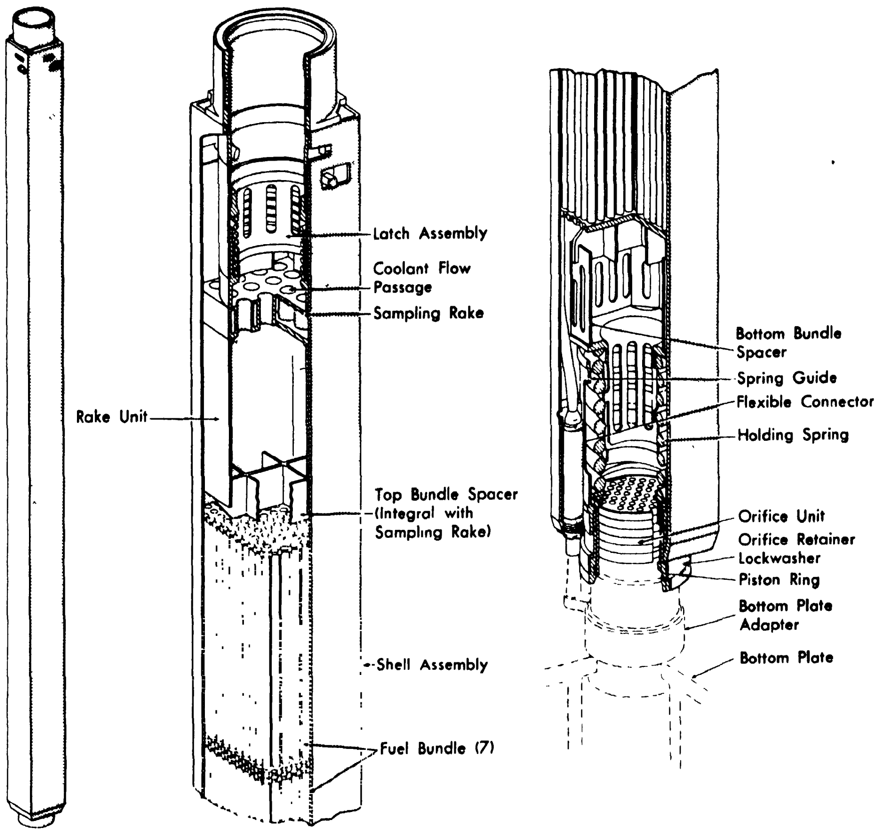


FIG. 4–14. Blanket assembly.

oxide region in the stacked bundles is $70\frac{3}{4}$ in., which corresponds to the active length of seed fuel. The length of uranium oxide, excluding all zirconium between bundles, is $64\frac{3}{4}$ in.

A complete blanket assembly consists principally of a latch assembly at the top, a sampling rake, the fuel bundles, and a spring and orifice system near the bottom. All these components are enclosed in an open-ended Zircaloy shell of square cross section with rounded corners. The blanket assemblies are held in the core cage between the bottom support plate and the top grid. This is accomplished by the spring loaded orifice resting against the adapter on the bottom support plate and the latch engaged to the underside of the top grid. The arrangement is such that, by use of remote handling equipment, an entire assembly can be unlatched from the top grid and removed from the core. A holding spring at the lower end assures firm seating when the assembly is in place and allows for manufacturing tolerances and differences in thermal expansion.

Because axial burnup of fuel is not uniform, provisions are included in the design for the rearrangement of fuel bundles within the assembly, with the least amount of mechanical difficulty, to achieve uniform over-all burnup. This is accomplished by loading all components except the orifice system from the top of the assembly and mechanically locking them in place. This scheme also permits reuse of the shell and nonfuel components.

During operation, the coolant enters the unit through the orifice system at the bottom; flows upward past the fuel elements, absorbing the heat they produce in the fission process; then flows through tubular passes in the coolant sampling rake and out through the top of the shell into the upper plenum chamber.

The sampling rake draws a portion of the coolant and directs it through tubing out through the bottom of the blanket assembly and, eventually, to a monitoring station outside the reactor vessel. This station, used to determine whether radioactive particles have been released to the coolant stream, is discussed in Chapter 8, and the associated piping and components are discussed in Chapter 12. Each blanket assembly contains a coolant sampling unit; in this way it is possible to detect and locate failed fuel elements in the blanket of the core. This provision is not extended to the seed fuel elements because of their higher integrity, established by past experience.

The over-all length of the assembly is $109\frac{7}{8}$ in. The active region begins $17\frac{3}{4}$ in. above the bottom support plate. Space is thus provided for the orifices and for the proper transition of the flow from circular to square cross section before it enters the fueled region. There is a similar space 24 inches long above the fuel region to accommodate the latch assembly and rake unit.

The approximate weight of one complete assembly is 510 pounds, apportioned as follows:

Fuel (UO_2)	284 lb
Zirconium alloy	186 lb
Stainless steel	40 lb

The basic active element of the blanket assembly is the fuel rod, consisting of cylindrical pellets of UO_2 enclosed in Zircaloy tubing. End caps, also of Zircaloy, are fusion-welded to the ends of the tube to retain the pellets in place and allow for attachment of tube sheets. Zircaloy-2 is used for this purpose because it is resistant to corrosion and because of its low tendency to capture neutrons. The composition of the Zircaloy is as follows:

Zirconium	98.3%
Tin	1.45%
Nickel	0.05%

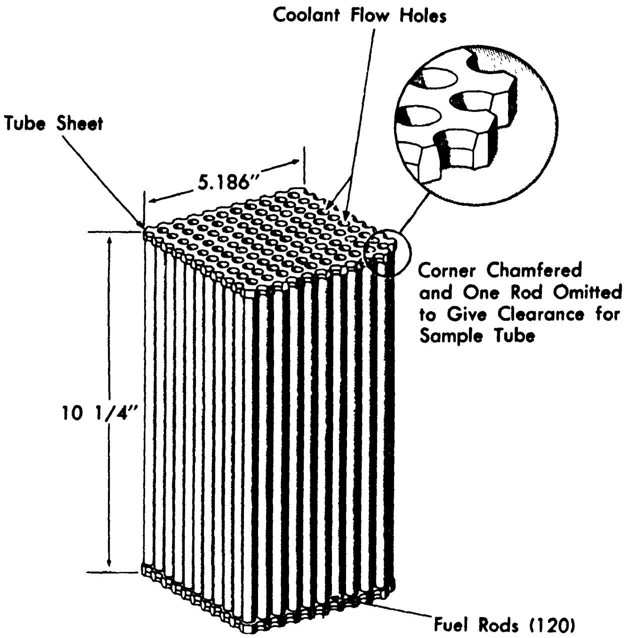


FIG. 4–15. Blanket bundle.

The assembled fuel rod, including end caps, is $10\frac{1}{4}$ in. long and approximately 0.411-in. OD. This length minimizes hydraulically induced vibration and the amount of radioactive material entering the coolant stream if an element or rod should fail. The 0.411-in. outside diameter satisfies thermal and nuclear considerations in regard to heat-transfer surface, heat flux, fuel centerline temperature, and metal-to-water and uranium-to-water ratios. The short rod length selected substantially increases the restriction to flow in the assembly because of the number of tube sheets. This is not a problem, however, because further restrictions must still be added through orificing to achieve the desired flow rates in the blanket.

A blanket bundle (Fig. 4–15) contains 120 fuel rods, arranged in an 11-by-11 square lattice. One fuel rod is omitted from one corner of the lattice to allow space for the sampling tube which transmits coolant from the sampling rake to the bottom of the assembly. To space and support the tubes, they are welded at the ends into drilled end plates called tube sheets. Between the ends of the rods the tube sheets are drilled with flow holes in a 12-by-12 square arrangement, including half edge holes. One corner of each tube sheet is cut away to allow space for the coolant sampling tube. A complete bundle is approximately $10\frac{1}{4}$ in. long and $5\frac{3}{16}$ in. square at the tube sheets.

The shell, which houses and supports all the other components of the blanket assembly, also serves as a conduit to contain and transfer the coolant from the inlet to the outlet plenum of the pressure vessel. It is a Zircaloy tube 5.6 in. square in cross section and 105¼ in. long, with a wall thickness of 0.20 inch. The shell is constructed so that all components may be removed, and the shell itself may be reused with new components.

The orifice unit, shown in Fig. 4–14, is mounted at the inlet or lower end of the blanket shell for two reasons: (1) in this location, the orifice tends to minimize the effects of any boiling that might occur during a transient; and (2) located at the bottom, it may be removed without affecting the latch assembly. Each blanket assembly contains one orifice unit. The orifice units restrict flow in the blanket assemblies to balance their flow rates with the seed flow rate.

Specifically, the orifice unit regulates the flow rate in each blanket assembly to obtain maximum power production without exceeding the maximum allowed metal surface temperature. Since power production in a blanket assembly varies according to the region in which the assembly is located, the restriction to flow differs for each region to provide the proper distribution of flow.

In the blanket of the core there are four such regions, as shown in Fig. 4–13. Region 1 consists of the 21 blanket assemblies comprising the center of the core; since they are separated from the seed assemblies, the flux in this region is comparatively low. Regions 2 and 3 comprise the blanket assemblies adjacent to the seed assemblies on the inside and outside, respectively; the thermal neutron flux in these regions is comparatively high. There are 24 assemblies in Region 2 and 40 in Region 3. Region 4 consists of 28 assemblies at the periphery of the active portion of the core.

As the lifetime of the core increases, there is a gradual rise in the ratio of power produced in the blanket to that produced in the seed. This is due to the production of plutonium, which increases the concentration of fissionable atoms in the natural uranium or blanket region of the core. To compensate for this power shift, the orifice units are made so as to be readily replaceable. As a result, the flow distribution to the blanket regions can be changed to accommodate the power shift after certain periods of operation.

The use of several multihole orifice plates in series is required to achieve a high pressure drop without using excessively high coolant velocities through the openings. The holes in adjacent plates are not in line, giving a staggered flow path that aids in obtaining the desired loss of pressure head.

The orifice unit with its housing is free (except for the restraint of the holding spring) to move longitudinally within the retainer for a distance

of about $1\frac{1}{4}$ in. As Fig. 4-14 shows, the seat of the holding spring bears on the top plate of the orifice unit. When the blanket assembly is out of the core cage, the thrust of the holding spring forces the unit to its lower limit of travel, where the lower surface of the flange bears on the orifice retainer. When the blanket assembly is installed in the core, the lower end of the orifice housing seats on the top surface of the bottom support plate adapter, forcing it upward within the retainer compressing the holding spring. The lower end of the retainer fits over the upper end of the bottom support plate adapter, and piston rings seal the space between the retainer and the adapter, minimizing coolant leakage from the inlet end of the assembly to the inactive spaces between the assemblies.

Spacers at the lower and upper ends of the bundle stack position the fuel bundles relative to the other components of the assembly. As shown in Fig. 4-14, the bottom bundle spacer supports the bundle stack and transmits the upward thrust of the holding spring to hold the bundle stack firmly in place.

The top bundle spacer is an integral part of the sampling rake unit mentioned previously and described below. At the lower end it is similar to the bottom bundle spacer and provides the bearing surface for the top of the bundle stack. The sampling rake at the upper end bears against the latching assembly to set the position of the top of the bundle stack.

The sampling rake is a hollow square box of stainless steel, $1\frac{1}{2}$ in. deep, which forms the upper end of the top bundle spacer. Sixteen short tubes pass through it, and the bulk of the exit cooling water passes through these tubes. The lower face of the box has holes in it which allow a small sample of the exit water to flow into the interior of the rake. The number, spacing, and size of the holes are such that the sample of coolant removed from the stream represents the entire cross section of the assembly. A short pipe extends into the interior of the rake so that the sample can be drawn off and transmitted through tubing and flexible connectors to the bottom of the assembly.

The latch mechanism which forms the upper end of the blanket fuel assembly is designed so that it can be easily removed from the assembly to allow replacement of the internal components. The assembly can be latched into the notches on the underside of the top grid by remote manipulation and easily unlatched by the same means when the blanket assembly is to be removed from the core. Each blanket assembly has a secondary lock to prevent the latch from rotating out of the notches in the grid, and thus ensure that the blanket assemblies cannot be lifted out of the grid by upward hydraulic forces if the bottom spring should fail.

The lower end of the blanket assembly is aligned and supported in the core cage by an adapter on the bottom support plate. All coolant entering the blanket assembly flows through this adapter. Each fuel assembly, in-

cluding the seed, has an adapter for its lower support. Basically, all the adapters are alike, although they differ somewhat in construction, both internally and externally, to satisfy the requirements of flow measurement and of the transmission line connecting the sample rake to the outside of the pressure vessel.

Each adapter is a cylindrical unit made of type-304 stainless steel. The outer surface of the upper portion is clad with stellite to resist wear when the piston rings in the lower portion of a fuel assembly rub against the adapter. The adapter is secured to the bottom support plate by means of a cylindrical nut and a thin-walled cup lock washer.

Seed assembly. A seed assembly, Fig. 4–16, consists of uranium-bearing fuel plates welded into integral clusters, supported at both ends by fittings somewhat similar to those used with the blanket assemblies. Each plate is a sandwich type of structure with a central region of highly enriched metallic uranium (approximately 90% U^{235}) alloyed with Zircaloy. The uranium alloy is clad with a 0.015-in. layer of Zircaloy-2, and the long edges are made with a flange that acts as a spacer between adjacent plates when they are assembled. The complete plate is $72\frac{3}{8}$ in. long and $2\frac{1}{2}$ in. wide. Its cross section is I-beam shaped, with a web thickness of 0.069 in. Fifteen of these fuel plates, along with two Zircaloy end plates, are welded together to form a boxlike structure, approximately $2\frac{1}{2}$ in. square in cross section, called a seed subassembly. Since the fuel plates and end plates are flanged, they form a series of parallel coolant passages approximately 0.069 in. between plates and $2\frac{1}{4}$ in. wide, running the full length of the plates.

The thickness of the seed fuel plates and coolant channels was selected on the basis of thermal design and manufacturing considerations. Because the heat flux and flow requirements are decreased with thinner fuel plates and the mechanical hot channel factors and flow friction factors are increased, there is a combination of fuel and channel thickness which requires the least hydraulic power to remove a given amount of heat. Analysis of these conditions indicated that the most favorable thickness for plates and channels, from a heat-removal standpoint, is about 0.060 in. This same study further indicated that very little difference existed between 0.060 and 0.070 in. To make manufacturing easier and decrease the cost by reducing the number of pieces, the high side of the thickness range was selected.

A complete fuel section or seed cluster consists of four seed subassemblies, welded together with half-inch spacers, as shown in Fig. 4–17. This forms the cross-shaped channel through which the control rod travels with sufficient clearance to allow coolant flow of approximately 8 fps to cool the control rod. An orifice plate located at the bottom of the channel controls the flow past the control rod. The finished fuel section is $5\frac{1}{2}$ in. square

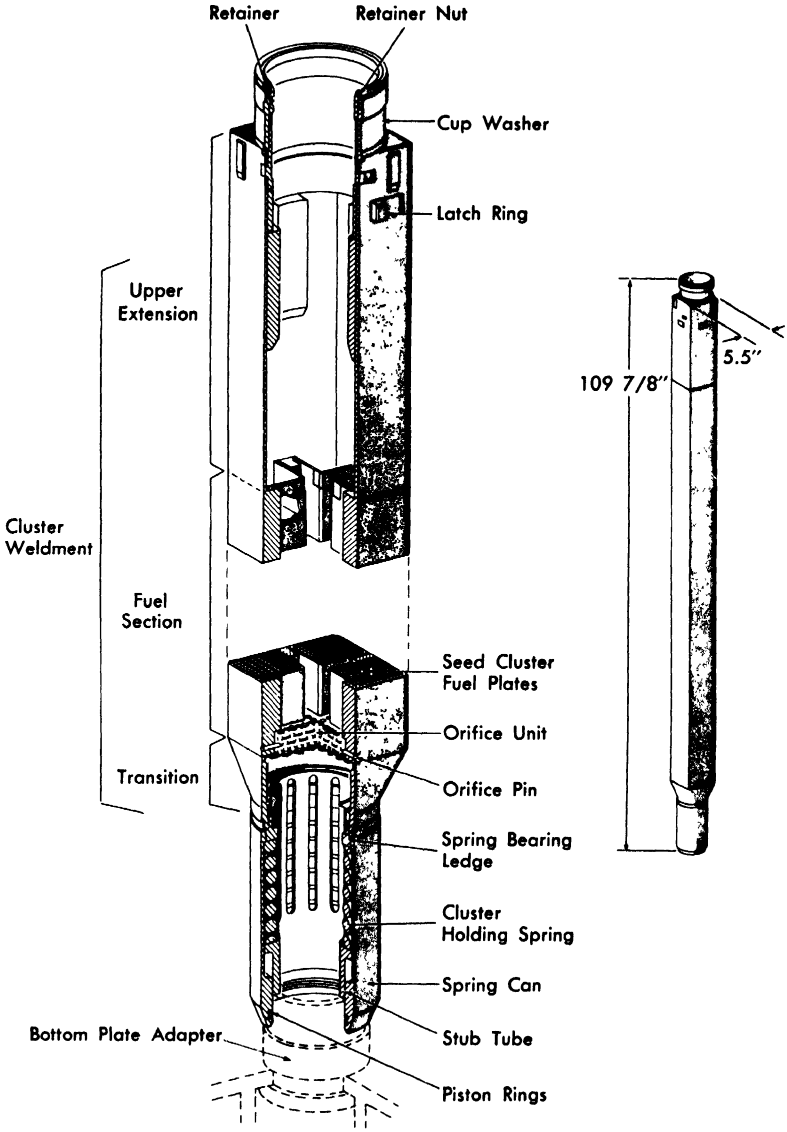


FIG. 4–16. Seed assembly.

and approximately $76\frac{1}{8}$ in. long. The active length of fuel is $70\frac{3}{4}$ in.—the same as in the blanket assemblies. When both seed and blanket assemblies are installed in the core, their active portions are on the same level.

Four of the seed clusters each contain, in one of the spacers, a 100-curie polonium-beryllium capsule to produce a suitable supply of neutrons for reactor startup.

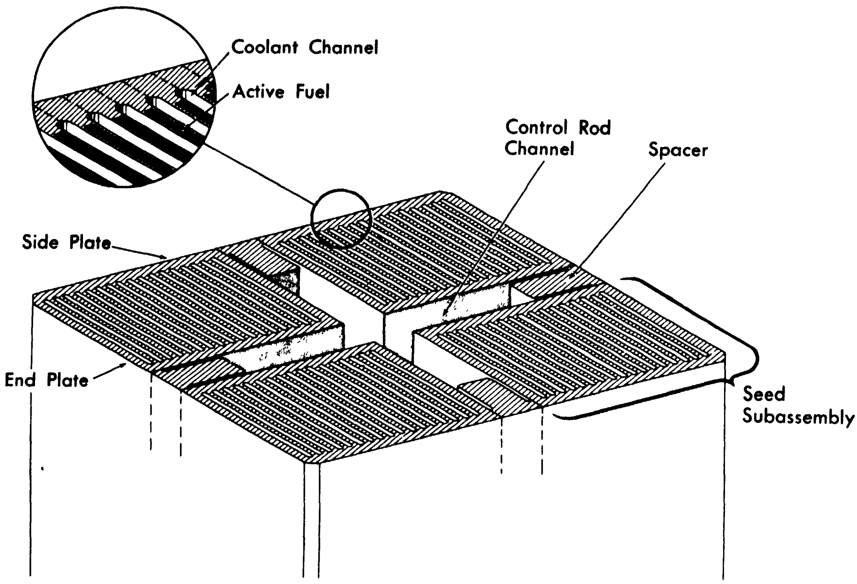


FIG. 4–17. Seed fuel cluster.

Unlike the blanket assembly, a seed assembly has no separate housing or shell as such. Instead, the flanges of the fuel plates, plus the end plates, when welded together form a complete assembly; the outer surfaces are machined to the final external cross-sectional shape.

A Zircaloy transition piece is welded to the lower end of the fuel section. The upper end of the transition piece is square in cross section, with the same external dimensions as the fuel section, while the lower end is cylindrical. A stainless steel spring can assembly is threaded to the bottom of the transition piece as shown in Fig. 4–16. This spring can assembly supports and aligns the seed cluster on the bottom support plate adapter and supplies a spring force to hold it in position between the bottom support plate and the top grid of the core cage.

A square tubular extension containing the latch mechanism is welded to the top of the seed cluster. It is the same size and shape as the outer walls of the cluster and extends $17\frac{3}{8}$ in. above the top of the fuel plates. As in the case of the blanket, the latch mechanism engages notches on the underside of the top grid of the core cage when the seed is installed. The notches on the underside of the grid prevent rotation of the latch under the action of the bottom holding spring, and a secondary lock prevents rotation of the latch if disengaged from the notch by downward forces transmitted through the control rod. The lower end of the control rod shroud is used to retain the latch ring for this purpose.

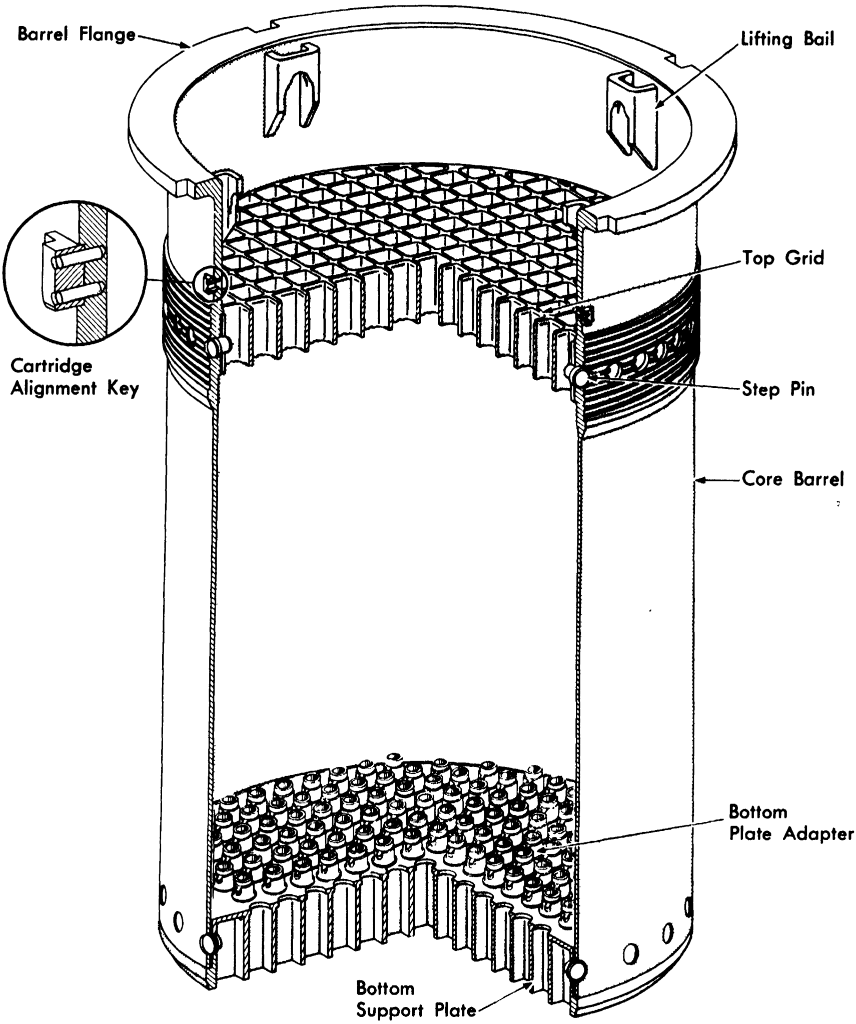


FIG. 4-18. Core cage.

Core cage. The core cage, Fig. 4-18, is the support structure for the blanket and seed assemblies. It holds the fuel elements in position, makes refueling by removal and installation of individual fuel assemblies possible, and assures correct alignment for operating the control rods. In addition, it contains and supports the connecting tubing of the blanket failed element detection and location system, the flow measurement instrumentation, and the inlet plenum water temperature thermocouples. The core cage transmits the weight of the core to the support ledge of the reactor

vessel. During operation, it transmits the hydraulic thrust of the coolant to the hold-down barrel in the upper part of the vessel.

The core cage has three principal components: a cylindrical barrel, a bottom support plate, and a top grid. It is made of type-304 stainless steel, is $108\frac{1}{2}$ in. in over-all diameter and $160\frac{5}{32}$ in. high, and weighs about $15\frac{1}{2}$ tons. The bottom support plate is cooled by inlet coolant flow, the top grid by exit flow. The barrel is cooled by water flowing downward in the space between the barrel and inner thermal shield of the vessel.

The core cage barrel contains and supports the bottom support plate and top grid. It is a cylinder approximately 156 in. high with an outside diameter, excluding the flange, of $94\frac{3}{4}$ in. Its wall thickness is 1 in. over the bottom portion and $1\frac{5}{8}$ in. near the top. The flange at the top of the barrel rests on the core support springs of the reactor vessel which, in turn, transmit the gravity load to the core support ledge of the vessel. The flange is 3 in. thick and $8\frac{1}{2}$ in. wide, with an outside diameter of $108\frac{1}{2}$ in.

Near the upper end of the barrel is a belt of enlarged diameter 18 in. high. A series of 23 grooves, $\frac{1}{2}$ in. wide and $\frac{1}{4}$ in. deep, separated by lands $\frac{1}{4}$ in. wide, is machined in the outer surface. The lands make a close fit (radial clearance 0.050 in.) against the inner thermal shield of the reactor vessel. This clearance was selected as a minimum to satisfy manufacturing and assembly considerations. The grooves form a labyrinth seal that prevents excessive quantities of coolant flow from bypassing the core through the space between the barrel and the shield. With a fixed clearance, the geometry of the seal is such that it offers the maximum obtainable resistance to flow for this type of seal. During operation, approximately 2% of the total reactor coolant flow leaks through this seal.

The barrel was made from two thicknesses of plate rolled and welded to cylindrical shape. A flange forging with a mating cylindrical portion was welded to the upper part of the barrel. Final dimensions were achieved by machining.

The top grid of the core cage maintains the upper alignment and spacing of both the seed and the blanket assemblies. It also carries the excess spring and hydraulic thrust of each fuel assembly. At the same time, its construction is such that an individual fuel assembly, after being unlatched, may be withdrawn through the corresponding opening in the grid. As Fig. 4–18 shows, the grid consists of an outer cylindrical ring, traversed by webbing arranged to form parallel rows and columns of square open-ended boxes or cells, spaced apart on 6 in. centers and with a web thickness of $\frac{9}{32}$ in. between openings. There are 145 openings in all, one for each fuel assembly. Where the webbing joins the inner surface of the ring there are additional openings of irregular shapes. These openings support and accommodate the instrument conduits passing from the bottom of the core to the top of the pressure vessel. The top grid has an

over-all diameter of about $91\frac{1}{2}$ in. and a height of 12 in. The openings in the top grid are all 5.726 in. square, providing sufficient clearance for withdrawing or inserting a fuel assembly, yet sufficiently close to ensure accurate alignment. Notches on the underside of the grid receive the latch bosses of the fuel assemblies when they are installed, locking them securely against rotation. The top grid and cage barrel are fastened together by 40 radial shear pins, made of Haynes-25 alloy, 2 in. in diameter, equally spaced around the central horizontal plane of the grid.

The bottom support plate is the weight-bearing component of the core cage assembly. At no flow, it supports both the weight and the additional downward thrust of the springs of the fuel assemblies and transmits this load to the barrel, which in turn transmits it to the reactor vessel. During operation, flow pressure differential produces an upward hydraulic thrust, relieving the major portion of the load from the bottom support plate. The resultant upward force of the core is taken up by the top grid. The bottom support plate, along with the bottom support plate adapters, provides a means for maintaining proper alignment of the fuel assemblies and also serves as a mounting for components of the core instrumentation system.

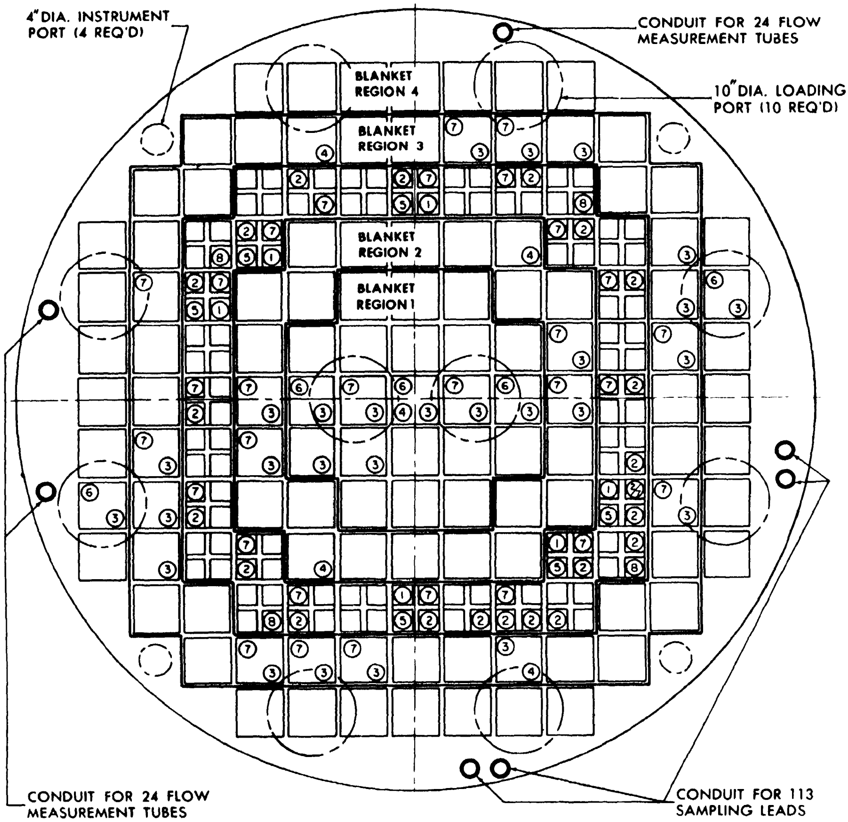
The stainless steel bottom support plate consists of an outer cylindrical ring $92\frac{13}{32}$ in. in diameter and 14 in. high, enclosing a lattice of vertical webbing. This lattice forms a pattern of square openings that correspond in alignment and position to the lattice openings in the top grid. To the upper surface of the lattice is attached a cover plate drilled with 145 holes, four in. in diameter, with centers corresponding to the centers of the square openings. The bottom support plate adapters, which align and support the fuel assemblies, are mounted in these holes.

The lower outer surface of the bottom support plate has a flange $4\frac{1}{4}$ in. high which bears against the bottom of the core barrel. A set of 28 radial shear pins, also made of Haynes-25, 2 in. in diameter and equally spaced in a horizontal plane attach the bottom support plate to the core barrel.

Both the top grid and bottom plate were made from a weldment of extruded cruciform shapes.

4–2.9 Core instrumentation. The instrumentation in the PWR core has a dual purpose: to provide the data needed for monitoring operations, and to furnish information that will be useful for future designs. There are three major classes: flow measurement instrumentation (FMI), temperature measurement instrumentation, and failed element detection and location (FEDAL) instrumentation.

Figure 4–19 shows the general layout of the instrumentation used with the core. Note that not every fuel assembly is instrumented. In part,



LEGEND

1 - CLUSTER SUBASSEMBLY CONTAINING SEED FUEL PLATE THERMOCOUPLES (6)

2 - POSITION OF SEED EXIT WATER THERMOCOUPLE (22)

3 - BLANKET ASSEMBLIES INSTRUMENTED FOR EXIT WATER TEMPERATURE (27)

4 - INLET WATER THERMOCOUPLE, BOTTOM PLATE (5)

5 - INLET WATER THERMOCOUPLE, SEED CLUSTER (6)

6 - FLOW NOZZLE, BLANKET REGION 1 (3) } (5)
FLOW NOZZLE, BLANKET REGION 4 (2)

7 - MODIFIED VENTURI, BLANKET REGION 1 (2) }
MODIFIED VENTURI, BLANKET REGION 2 (4) } (31)
MODIFIED VENTURI, BLANKET REGION 3 (9) }
MODIFIED VENTURI, SEED REGION (16) }

8 - SEED CLUSTER CONTAINING SOURCE (4)

ALL BLANKETS HAVE SAMPLING RAKES FOR FEDAL SYSTEM.

FIG. 4-19. Core instrumentation layout (presented on top view of core).

this was dictated by space limitations; it was necessary, for example, to keep tubing and conduits clear of fuel-handling operations in the reactor vessel. Accordingly, instrumentation has been located on assemblies selected for their representative positions. The locations of the FEDAL sampling connections are not shown in Fig. 4-19. As described earlier in this chapter, there is one for each blanket assembly, none being required

for the seed assemblies. The remaining instrumentation can be summarized as follows:

Flow measurement instrumentation (FMI). The flow measurement system consists of differential-head type meters (modified venturis and flow nozzles) mounted below individual fuel assemblies as part of the bottom support adapters. The pressure signals are transmitted by tubing from the meter through the reactor vessel head and to differential pressure cells mounted on the head. To avoid interference with remote refueling, the pressure transmission leads are confined to the space available below and around the periphery of the core. Further, to facilitate assembly, the tubing is divided into three sections: a lower section extending radially outward from the fuel assembly and vertically along the core structural supports to the top grid, an upper section from the top grid to the vessel head, and an external section of high-pressure tubing to the differential pressure cells. Because of space limitations, a special seal was required to connect the sections. The pressure tubing of each section was connected in tube sheets; tubes were connected between adjoining tube sheets by short flexible lengths of thin wall tubing. A threaded connection and a small diameter gold ring completed the seal between each end of each short flexible tube and its respective tube sheet.

In placing the flowmeter in the bottom support adapter, standard installation procedures (8 pipe diameters upstream and 4 pipe diameters downstream) could not be followed, since space was not available. For this reason, it was necessary to verify by full-scale tests that the flow conditions upstream and downstream from the meter would not affect its operation. These tests also served to calibrate the flowmeters. As shown in Fig. 4–19, 36 flow meters are provided, 16 in the seed and 20 in the blanket.

Temperature measurement instrumentation. The thermocouples for temperature measurement are chromel-alumel wires separated and insulated by zirconium oxide and contained within stainless steel sheaths. The primary pressure seal at the pressure vessel head is accomplished by a Nicro-brazed joint between the stainless steel sheath and a stainless steel plug structurally welded into the vessel head. Above this joint, a secondary mechanical seal for each chromel and alumel wire is employed. Between the primary and secondary seal, a joint is made between the thermocouple leads and larger diameter wires which are connected to cabling and read-out instrumentation.

Temperature measurements at the centerline of the seed fuel plate and at the inlet and outlet of the core are provided. Six of the seed assemblies have fuel plate and inlet water thermocouples, and 16 of the seed assemblies have outlet water thermocouples. In the blanket region, thermocouples measure the outlet temperature of 27 assemblies. In addition,

there are five inlet water thermocouples mounted in the bottom support plate, as shown in Fig. 4–19. In each of these, the sensing bulb is located approximately 4 in. below the top surface of the bottom support, to sense the temperature of the water entering the associated blanket fuel element.

Because of the difficulties encountered in fabricating and installing seed fuel plate and seed inlet water thermocouples, some of the units were damaged and the readings are not considered reliable. It was found that the aging process employed to stabilize thermocouple calibration introduced local areas of brittleness in the sheathing of the 0.040-in.-diameter seed plate and seed inlet thermocouples by precipitation of carbon along the grain boundaries. This brittleness, in combination with the mechanical handling required during installation, caused minute local fissures, through which system water later leaked and caused thermocouple failure. However, the five other inlet thermocouples and the 49 outlet thermocouples are of a larger size and did not require such complicated installation methods. These units are operating satisfactorily.

Failed element detection and location (FEDAL). The design of the fuel element detection and location system provides a means of (1) obtaining a representative sample of each blanket assembly effluent, (2) positively identifying the sample with the assembly from which it originates, and (3) transporting the sample through the pressure vessel wall to the monitoring station with minimum leakage at the various system junctures.

An assembly can be identified with its effluent sample by leading an individual sampling tube from each of the 113 blanket fuel assemblies to a multiport sampling valve on the closure head. From the multiport valve, the effluent is ducted to a monitoring station, where its activity is measured by means of a delayed neutron monitor.

4–2.10 Control rods and accessories. Each of the 32 seed assemblies has a control rod and a rod drive system. In response to manual or automatic controls, the drive system moves the rod vertically down into or up out of the seed assembly as required by the need for changes in reactivity such as occur during startup and shutdown. The drive mechanisms are in cylindrical housings attached to the top of the closure head of the reactor vessel, directly above the seed assemblies they control.

With this hole penetration pattern and size, $3\frac{5}{8}$ in. is the largest control rod span to meet code requirements for the reactor vessel head, permit removal and replacement of the control rods through openings in the head, and to allow use of a one-piece shroud.

Accessories used with the control rods, as shown in Fig. 4–20, are the rod drive mechanisms, including rod position indication sensing elements, and the shaft and shroud assemblies. The shaft and shroud assemblies

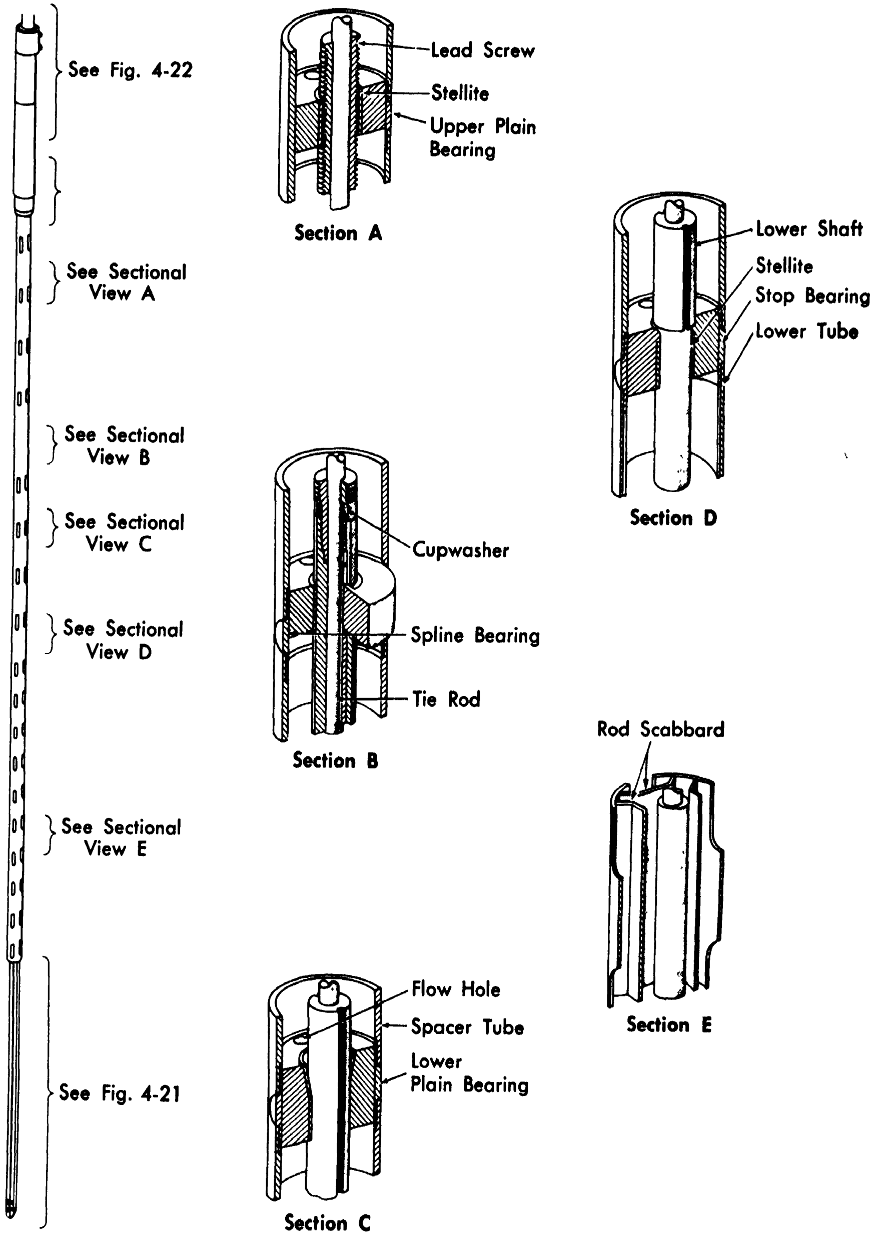


FIG. 4–20. Control rod and accessories.

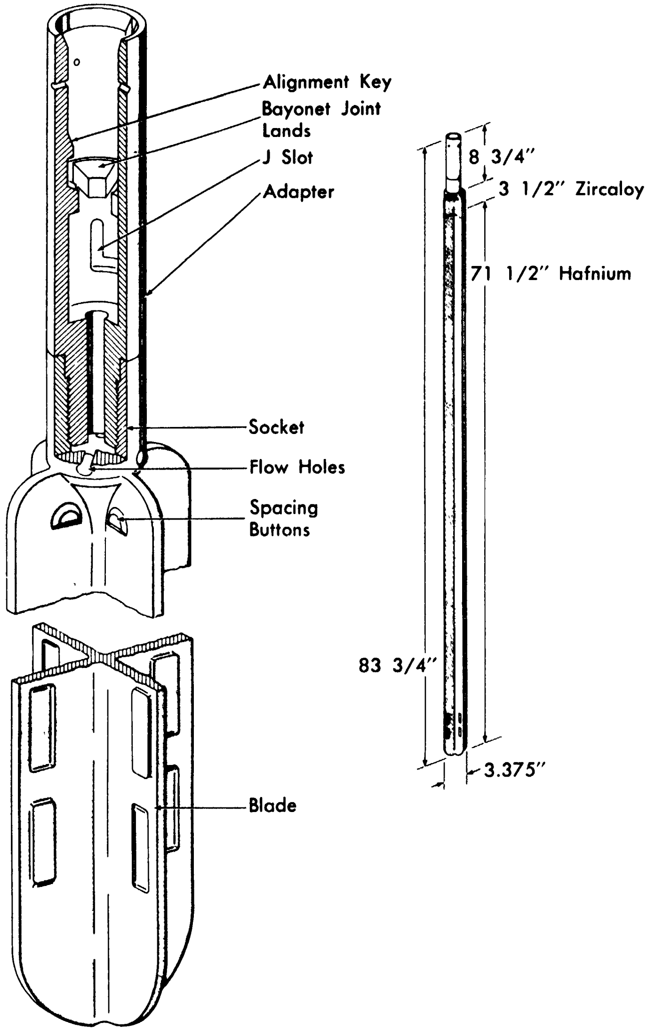


FIG. 4–21. Control rod and adapter, over-all and sectional views.

extend downward from the mechanism ports in the closure head of the reactor vessel, go through the top grid of the core cage, and enter the tops of the seed fuel assemblies. Here each shroud seats within the assembly latch mechanism. The shaft is connected to the control rod by means of a device that permits uncoupling by manipulation from above the head of the reactor. Thus the control rod can be left fully inserted in the seed when the drive mechanism and shafting are removed.

Control rod assembly. There are 32 identical control rods, one for each seed cluster. As Fig. 4–21 shows, each consists of two principal parts: the

blade, and an adapter at the upper end through which the rod is connected to the lead screw of the drive mechanism.

The rod, $83\frac{3}{4}$ in. long, is cross-shaped in section, with a span of $3\frac{3}{8}$ in. The plates of which it is formed are 0.226 in. thick and are welded at right angles to each other to form a continuous unit, as shown in Fig. 4–21. The lower $71\frac{1}{2}$ in. of each rod consists of hafnium; the remainder is Zircaloy. The length of the hafnium portion corresponds to the active length of the fuel in the seed cluster. Hafnium is used for this purpose because it is a strong absorber of thermal neutrons and has satisfactory mechanical properties. Near the lower end of the rod, spacing shoes are stamped in the rod to assure clearance for a cooling channel in case the rod should be forced against a channel wall. There are bosses or buttons welded to each face near the upper end of the rod proper. These buttons assure proper alignment when the drive shaft is remotely connected to the control rod.

A threaded socket welded to the top of the control rod is used to attach the lower end of the adapter to the upper end of the control rod (Fig. 4–21). Small holes, drilled obliquely in the lower end of the socket, allow coolant to flow upward through the adapter to cool the lower end of the tie rod through which the control rod is attached to the drive shafting of the shroud assemblies. This adapter is used for remotely connecting and disconnecting the control rod from the drive shafting. As a result, the control rod can be kept fully inserted in the seed assembly during shipping and installation, then attached to the rod drive shafting for controlled withdrawal after the seed assembly is ready to put into service.

Control rod shaft and shroud assembly. The control rod shaft and shroud assembly (Fig. 4–20) receives the axial motion generated by the control rod drive mechanism and transmits it to the control rod to position it in accordance with power level requirements. Its major components are the shroud, a flange block and its accessories, a series of four bearings for support and alignment of the drive shafting, the drive shaft, and the tie rod.

The shroud assembly serves several purposes: it provides bearing support and alignment for the shafting, it protects the shafting against the lateral thrust of the coolant flow, and it locks the upper end of the seed assembly into place in the top grid. The shrouding extends from the mechanism housing in the vessel head to the inside of the upper end of the seed assembly, a total length of $203\frac{5}{8}$ in. or about 17 ft. Since the diameter of the shroud is just under 4 in., it has been necessary to use every possible refinement of mechanical design to obtain sufficient rigidity over this length. The shroud must be installed through the mechanism housing, after the vessel closure head is in place but before the rod drive mechanism is installed.

The lower tube, which is 90 in. long, differs from the others of the shroud assembly in several respects: it has a thinner wall section, and is fitted

with internal flow baffles, as shown in Fig. 4–20. These baffles are arranged so that they form a cross-shaped passage for the control rod. The walls of the lower tube are pierced with long narrow holes that permit coolant to flow in and out through the space between the baffles and the wall of the tube. When the control rod is fully withdrawn from the seed assembly, it is housed entirely within the lower tube. A tapered slot cut into the bottom end of the lower tube engages an elongated pin on the upper end of the seed assembly, latching it in place.

The drive shaft is a hollow stainless steel (type 17–4 PH) tube through which the motion of the rod drive mechanism is transmitted from the mechanism lead screw to the control rod. Not including the lead screw, of which it is a continuation, the drive shaft is approximately 162$\frac{5}{8}$ in. long. The upper part is 1$\frac{3}{8}$ in. OD, the lower portion 1$\frac{1}{8}$ in. OD. The internal diameter is approximately $\frac{11}{16}$ in. throughout, and accommodates a $\frac{5}{8}$-in.-diameter tie rod by means of which the drive shaft is connected to the control rod during installation. When assembly is completed, the bottom 2$\frac{5}{8}$ in. of the drive shaft is enclosed within the top end of the control rod adapter. The tie rod is so constructed that remote engagement and disengagement to the control rod can be accomplished by working with special tools from above the drive mechanism housing. The tie rod is approximately 261 in. long and extends up through the drive shaft into the drive mechanism. Near the upper end it is threaded and, when installed, is held in place by a tie rod nut with a stellite outer facing that draws it up firmly into engagement with the control rod adapter at the lower end.

Control rod drive mechanisms. The control rod drive mechanism, shown in Fig. 4–22, is a rotating nut and translating screw device that operates in the reactor coolant water and is driven by an induction motor in which the stator is isolated from the coolant water by a stainless steel liner or can. When the rod is to be moved, the stator is supplied with AC power at a frequency variable from 0 to 0.773 cps, which causes it to rotate at a speed of from 0 to 22 rpm. The phase sequence of the power applied to the stator can be reversed to give rotation in either direction for insertion or withdrawal of the rod. When the rod is to remain stationary, a DC holding voltage is applied to the stator. This voltage keeps the rotor stationary, preventing it from being driven by the gravity load of the control rod and shafting.

The rotor of the motor, which is exposed to the coolant water, drives transverse shafts on which are mounted two pairs of arms that pivot in a seesaw action. Ball-bearing mounted roller nuts are attached to the lower ends of these arms to drive the lead screw. The roller nuts are similar in function, but not in construction, to the split nut used to drive the tool

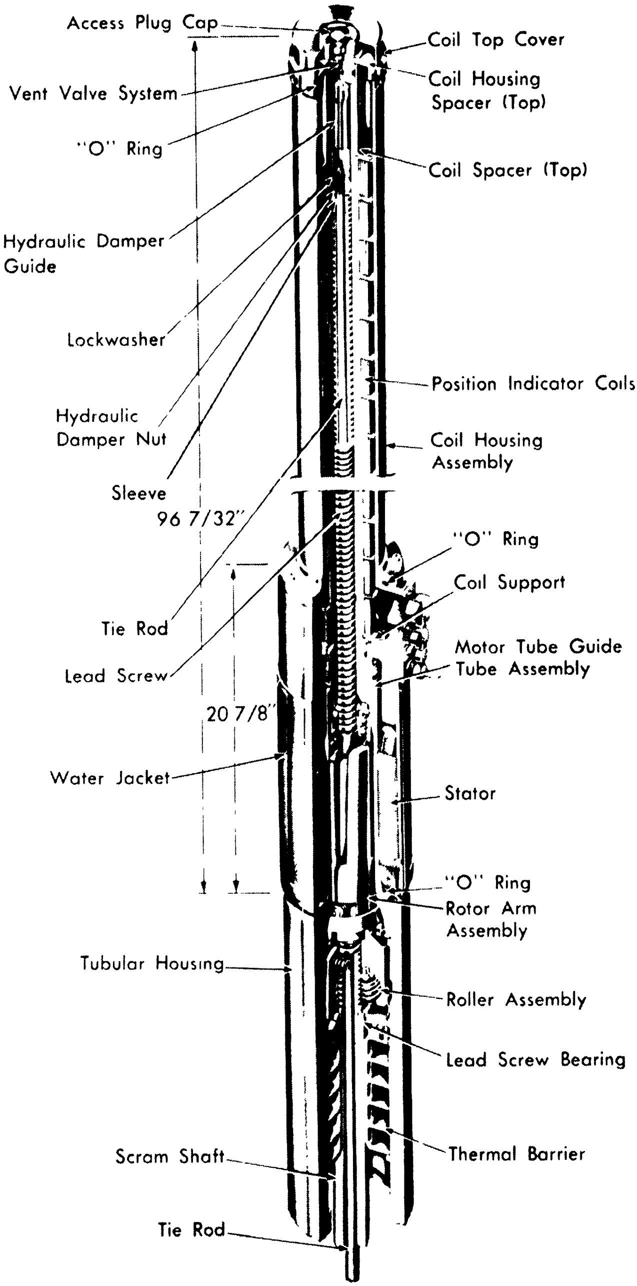


FIG. 4-22. Control rod drive mechanism.

rest of an engine lathe. When the mechanism is in normal operation, a magnetic force holds the arms in a closed position so that the lead screw engages the nuts. When a safety shutdown (usually called a scram) occurs, this magnetic force is cut off, disengaging the nuts from the lead screw, and the control rod is gravity driven to its lowest (full inserted) position within the seed assembly. Because of this construction, no special procedure is required to pick up the rod when normal operation is resumed.

To prevent impact damage, there is an enlarged diameter region, faced with stellite, on the tie rod lock nut (damper nut) in the mechanism, and a constriction in the lower end of the mechanism guide tube. When the rod approaches the end of downward travel, the enlarged diameter meets the constriction. Although there is a slight clearance, water flow past this narrow passage creates an effective damping action.

At the upper end of the tie rod, where it is connected into the rod drive mechanism, there is a short length of magnetic material. As the rod is raised, the magnetic material moves through a series of 24 differential transformer coils mounted one above the other, 3 in. apart on centers, increasing the energy transfer to each secondary as it passes through. A corresponding series of indicator lamps, connected to the secondaries, are thereupon lighted successively, showing rod position within approximately plus or minus $1\frac{1}{2}$ in.

A summary of control drive mechanism data is presented in Table 4–2.

TABLE 4–2

CONTROL DRIVE MECHANISM DATA

Over-all length of mechanism with lead screw withdrawn, inches	$115\frac{15}{32}$
Height of the mechanism above the stainless steel head penetration housings, inches	$96\frac{7}{32}$
Average inside diameter of mechanism housing, inches	$4\frac{1}{2}$
Length of stroke, inches	$70\frac{1}{2}$
Weight of rod drive mechanism, pounds	317
Weight of shaft and shroud assembly, pounds	285
Weight of control rod and adapter, pounds	55
Normal mechanism speed, inches per minute	11
Safety shutdown time from shutdown signal to release of rod, seconds	0.350
Safety shutdown time from shutdown signal to full insertion (starting with rod fully withdrawn), seconds (approx.)	1.45

Figure 9–4 shows the arrangement of the 32 control rod drive mechanisms and associated cabling and cooling piping mounted on the reactor

vessel head. Also shown is the instrumentation trellis which supports the differential pressure cells for the flow measuring instruments located within the reactor vessel. The equipment on the reactor vessel head was carefully arranged to provide access to the ten refueling ports without having to dismantle the trellis structure or any of the permanently installed piping. Except when removing fuel assemblies through the two center ports, no disassembly of mechanisms is required. Because of height restrictions, removal of assemblies from the two central ports requires disassembly of at least one mechanism to permit passage of the assembly while retaining sufficient water above it for shielding. Removal of any seed cluster requires disassembly of its associated control rod drive mechanism and accessories. All equipment in the pressure vessel head area was made waterproof to permit the equipment to remain in place during operations that require flooding of this area.

4-3. Core Thermal Capabilities

4-3.1 Design and operational limitations. The design of the reactor core was developed around two limiting criteria which, at the time design work was started, had experimentally been demonstrated to be a safe and suitable basis. As indicated in Chapter 1, these were defined as follows:

(1) The maximum metal surface temperature in the core should be limited to that associated with the onset of local boiling during steady-state operation. To meet this criterion with water at 2000 psi, the maximum metal surface temperature, T_{sm}, was limited to the corresponding saturation temperature of 636°F. This provided a margin of 6°F with respect to the 642°F temperature at which significant local boiling is predicted by use of established boiling correlations. In addition, core performance was evaluated under plant conditions which deviated from the nominal by 30 psi on reactor coolant system pressure and 5°F on core inlet coolant temperature.

(2) Bulk boiling should be avoided in the water leaving the hottest channel of the core during an operational accident such as loss of coolant flow. During the transient following such an accident, the bulk water temperature is permitted to increase to just below 636°F.

As the design of PWR progressed, experimental work at Westinghouse Bettis Atomic Power Division and other AEC installations indicated the extent to which these criteria might be transcended without endangering the core. The results provide a useful basis for evaluating the ultimate potential of the reactor core, particularly when used in conjunction with the core instrumentation, which provides actual power distribution as

compared with design values. The work published during the past two years has defined conditions under which burnout of fuel elements associated with film boiling might be expected. As a result, it is possible to define operational limits based on burnout ratios (ratios of burnout heat flux to actual heat flux) in order to yield a suitable safety factor.

In applying this information, attention must be given to other limitations that may apply, such as stresses and material properties. For example, early in core life, it is expected that the upper limit of PWR steady-state power will be that level at which the loss of coolant flow due to local boiling can be accommodated without causing the burnout ratio, at any time during the ensuing transient, to fall below a preselected value. Later in core life, however, a more severe limitation is faced: buildup of plutonium will cause higher heat fluxes in the blanket region adjacent to the seed. This means that the maximum permissible power will have to be limited to that value which has been found experimentally to be the threshold of failure from another cause. This threshold appears to be associated with the temperature at which centerline melting of the uranium oxide occurs. The value of this heat flux has been determined experimentally to be approximately 500,000 Btu/hr·ft^2 for fuel rods with undefective cladding and about 425,000 Btu/hr·ft^2 for rods with defective cladding. In this case a defective cladding is considered one which admits water into the void between fuel and clad.

The basis for evaluating the ability of the Shippingport reactor core to produce more power will be twofold. First will be the detailed operational determination of power characteristics of each of the orificed regions of the core as indicated by core instrumentation and comparison with design values. Second will be the relationship of operational heat flux and pressure drop characteristics to those experimentally determined in the laboratory to be limiting. Seed and blanket calculations to date have utilized burnout and local boiling pressure drop data obtained experimentally in rectangular channel test sections up to 27 in. in length. Tests on a 6-ft-long rectangular channel have been completed, confirming existing burnout and pressure drop correlations.

4–3.2 Power peaking data used in design. In evaluating the thermal capabilities of the reactor core, extensive use was made of the thermal neutron fluxes and associated power densities discussed in Chapter 3. These patterns show a nonuniform production of power throughout the core. Accordingly, limiting criteria are applied to those regions of the core which are subjected to the most adverse combination of power peaks in both the radial and axial directions. The power distribution patterns used for computing rated power for the core were those associated with equilibrium xenon conditions.

The radial neutron flux patterns have consisted of gross radial flux distributions as well as local peaks occurring in the coolant channels of the core and in adjacent seed and blanket fuel areas. Because of the moderating properties of the water in a local channel, more thermal neutrons (and therefore a higher thermal neutron flux) appear in this region. The largest water channels in the PWR are those between fuel assemblies and in the control rod channels when the control rods are withdrawn. The flux distribution in the water channels and adjacent fuel is calculated by point-by-point multiplication of the gross radial flux distribution and the appropriate water channel peaking factor for a given location. In utilizing the gross radial flux distributions, different proportionality constants must be applied to find the actual power generation in the seed and in the blanket, because of the dissimilar loadings of the seed and the blanket.

Axial distribution of thermal neutron flux vertically through the core varies with radial position, control rod programming, and core lifetime. When the control rods are positioned so that rods are either all the way in or all the way out, the axial neutron flux plot and hence the power generation pattern tend to have a cosine characteristic. When certain rods are partially inserted, the pattern is skewed toward the lower part of the core. The curves of axial power distribution used for design purposes were those for partial rod insertion and were estimated to have peak-to-average ratios of 2.00 for the seed and for blanket regions 2 and 3, and 1.77 for blanket regions 1 and 4. Both the radial and axial peaking factors are summarized in Table 4–3. In applying these factors, consideration is given to the fact that approximately 95% of all heat produced is released directly in the fuel and is due to the slowing down of fission fragments. The remaining 5% is released in the coolant water and structural materials and is the result of neutron moderation and alpha, beta, and gamma radiation.

In addition to the nuclear influences on power distribution, certain so-called "engineering hot channel factors" have influenced power peaks. These account for variations in flow distributions and variations in heat production or heat transfer brought about by manufacturing tolerances. In the Shippingport reactor, such variations have been categorized by their influence on *temperature rise of the coolant*, *heat generation rates*, and *temperature drop across the water film* adjacent to the fuel-element surface. The effects of the various types of fabrication inaccuracies are identified as hot-channel factors $F_{\Delta T}$, F_q, and F_θ, respectively, and are shown in Table 4–3. In use, analytical cases are postulated which describe the manner in which these variations can occur and which permit identification of their cumulative effects. As a description of a local event, they represent a multiplying factor applied to the average reactor conditions and thereby determine the extent of approach to design limits.

TABLE 4-3

PWR HOT-CHANNEL FACTORS AT BEGINNING OF CORE LIFE

BLANKET

		$F_{\Delta T}$	F_{θ}	F_q
Engineering				
Flow distribution		1.07	1.06	1.00
Fuel diameter		1.01	1.01	1.01
Fuel concentration (1.01 density variation, 1.002 concentration variation)		1.01	1.01	1.01
Meat eccentricity		1.02	1.03	1.03
Heat-transfer correlation		1.00	1.25	1.00
Pitch, bowing, and rod diameter		1.11	1.14	1.00
Absence of one fuel rod (for FEDAL System)		1.01	1.01	1.01
Cooling effectiveness		1.06	1.05	1.00
Differential orificing		0.87	0.89	1.00
	Product	1.16	1.49	1.06
Nuclear				
Maximum to average,* radial	Region 1	1.24	1.24	1.24
	Region 2	1.94	1.94	1.94
	Region 3	1.82	1.82	1.82
	Region 4	0.90	0.90	0.90
Maximum to average, axial	Region 1	—	1.77	1.77
	Region 2	—	2.00	2.00
	Region 3	—	2.00	2.00
	Region 4	—	1.77	1.77
Maximum to average, bundle depression	All regions	1.24	1.28	1.28
End cap peaking	All regions	—	1.18	1.18
Product	Region 1	1.54	3.33	3.33
	Region 2	2.40	5.85	5.85
	Region 3	2.25	5.49	5.49
	Region 4	1.11	2.41	2.41
Over-all hot-channel factors	Region 1	1.78	4.95	3.52
	Region 2	2.77	8.71	6.19
	Region 3	2.60	8.17	5.80
	Region 4	1.29	3.58	2.54

* Average of the entire blanket region.

TABLE 4-3 (*continued*)

SEED

	$F_{\Delta T}$	F_θ	F_q
Engineering			
Flow distribution	1.07	1.06	1.00
Meat thickness variation	1.05	1.13	1.13
Fuel concentration	1.02	1.04	1.04
Meat eccentricity	1.05	1.14	1.14
Channel thickness	1.10	1.11	1.00
Heat-transfer correlation	1.00	1.25	1.00
End conduction from fuel	0.95	0.95	0.95
End coolant channel	0.97	0.95	0.95
Product	1.22	1.77	1.21
Nuclear			
Maximum to average, radial	1.25	1.25	1.25
Maximum to average, axial	—	2.00	2.00
Maximum to average, water hole	1.29	1.29	1.29
Blanket end cap effect	—	1.09	1.09
Product	1.61	3.51	3.51

4-3.3 Influences of orificing on thermal capabilities. Because of the nonuniform pattern of power generation, most effective use of the coolant flow through the core was obtained by apportioning the flow through various regions of the core in accordance with the expected power production. As discussed under the section on fuel assembly arrangement in this chapter, the core was divided into five regions, with different orifices in each region. These regions are designated as the seed region and four blanket regions: Region 1, the interior of the core; Region 2, the blanket assemblies adjacent to the seed and interior to it; Region 3, the blanket assemblies adjacent to the seed and exterior to it; and Region 4, the blanket assemblies on the periphery of the core. The orifices are designed to limit flow through the assemblies in the regions of lower flux to the minimum needed to avoid local boiling at the hottest spot in each region.

In addition to the gross orificing, the flow holes in the tube sheets of each of the blanket bundles were made in three different sizes. They are proportioned to provide greater water flow near the edges of the bundles to lessen the effect of local heat generation peaks in that area. Experimental results on differential orificing show that while proper flow distribution occurs at the tube sheet, interior to the bundle the distribution returns to normal parallel flow. Peak heat generation, however, occurs at

the end caps in the vicinity of the tube sheet so that flow redistributed at the tube sheet is beneficial. Total reactor power is increased by a maximum of 6% through the differential orificing of blanket bundle tube sheets.

While the differential orificing associated with the bundle tube sheets is fixed, the orifices for controlling flow to the gross radial heat generation pattern are replaceable. Nevertheless, it was desired to size these orifices so that they would not have to be replaced during core life. At the start of core life, the power sharing between seed and blanket is such that about 47% of the power is produced in the seed and 53% in the blanket. However, as operations progress, Pu^{239} is formed in the blanket from the consumed U^{238} and, as a result, power output from the blanket is increased. This power shifting as a function of lifetime has been analyzed and the seed-blanket power sharing determined as a function of lifetime. In particular, core neutron flux distributions were prepared for three specific times over the estimated life of the core. This core lifetime can be identified in terms of control rod patterns as follows: (1) eight control rods out of 32 fully inserted, and the remaining 24 fully withdrawn; (2) four control rods fully inserted; and (3) no control rods inserted. For each of these conditions, two-dimensional flux distributions were determined on which to base the thermal and hydraulic design.

In establishing orifice sizes for the blanket regions, consideration had to be given not only to the pressure drop characteristics of core components and the orifices but also to the effect of plutonium buildup and associated power shift on the power capabilities of the reactor operating with a selected set of orifices. The pressure drop characteristics were estimated from calculations and experiments; the blanket-seed power sharing was determined from theoretical physics predictions. The predictions of power sharing for the three rod programs mentioned earlier are tabulated in Table 4–4, along with region gross radial flux factors with various control rod patterns. These gross radial factors are ratios of peak power in a subregion to the average power in the region.

TABLE 4–4

SEED-BLANKET POWER SHARING AND RADIAL FLUX FACTORS

No. of rods inserted	Power sharing		Gross radial factors by region				
	Seed	Blanket	1	2	3	4	Seed
8	0.464	0.536	1.24	1.94	1.82	0.90	1.25
4	0.417	0.583	1.18	2.09	1.96	0.89	1.26
0	0.403	0.597	1.21	2.53	2.34	0.76	1.21

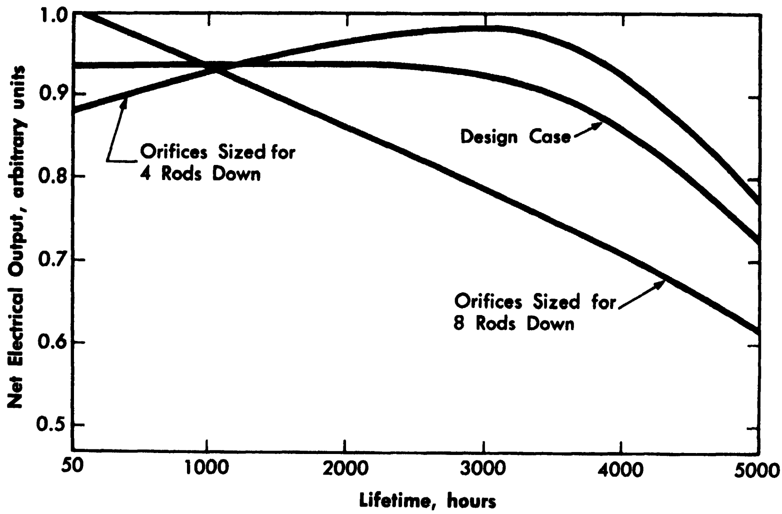


FIG. 4-23. Performance based on 636°F maximum surface temperature.

With the information on power sharing and radial factors, flow orificing may be designed so that maximum power is obtained at any particular time in core life. However, a choice of a time in core life to maximize power does influence power output over the rest of the life of the core.

Fixing the flow distribution to give maximum power for the condition of full insertion of eight rods has the effect of decreasing the allowable power as core life increases and plutonium builds up in the reactor. Similarly, if the coolant flow is fixed for maximum power when four rods are inserted, the power must be reduced early in core life when seed power is higher and again later in core life as more power is produced in the blanket. A flow distribution intermediate to the above two results in a power curve which is relatively constant over a major portion of core life. An intermediate flow distribution was selected which does not allow the maximum metal surface temperature T_{sm} to exceed 636°F. A comparison of the three orificing schemes is given in Fig. 4-23, using an arbitrary power scale.

With the selected orificing arrangement, the maximum fuel-element surface temperature in the seed region is 636°F early in seed life and drops progressively to about 594°F at the end of seed life. In Regions 2 and 3 of the blanket, adjacent to the seed, the maximum fuel-element surface temperature is 619°F at the beginning of seed life and increases progressively to 636 and 634°F, respectively, at the end of seed life. Regions 1 and 4 of the blanket have intermediate characteristics. It is expected that the core as presently constituted can operate with its first seed for a period equivalent to that required for the production of rated power for more than 3000 hr. On the basis of irradiation tests, it is estimated that

TABLE 4–5

THREE-LOOP BLANKET PERFORMANCE DATA

	Region			
	1	2	3	4
No. of assemblies	21	24	40	28
Output per assembly, avg, net Mw	0.25	0.40	0.28	0.15
Output of region, net Mw	5.31	9.67	11.32	4.20
Coolant flow per assembly, lb/hr $\times 10^{-4}$	8.24	17.00	15.80	6.07
Total flow of region, lb/hr $\times 10^{-6}$	1.73	4.08	6.32	1.70
Velocity, fps	6.21	12.81	11.91	4.57
Heat transfer coefficient, Btu/hr·ft^2·°F $\times 10^{-3}$	3.30	5.89	5.56	2.58
Heat flux, avg, Btu/hr·ft^2 $\times 10^{-4}$	5.20	8.29	5.82	3.08
Heat flux, max, Btu/hr·ft^2 $\times 10^{-4}$	19.5	34.3	32.2	14.1
Coolant temperature rise, avg, °F	36.9	28.6	21.5	29.7
Coolant temperature rise, max, °F	70.1	53.0	53.4	68.8

TABLE 4–6

THREE-LOOP SEED PERFORMANCE DATA

Output, net electrical, Mw	26.5
Coolant flow, lb/hr $\times 10^{-6}$	7.6
Velocity, fps	19.9
Heat transfer coefficient, Btu/hr·ft^2·°F	9,060
Coolant temperature rise, avg, °F	49.1
Coolant temperature rise, max, °F	96.3
Heat flux, avg, Btu/hr·ft^2	98,400
Heat flux, max, Btu/hr·ft^2	418,000

the blanket should be capable of lasting for at least the lives of two seeds. Table 4–5 shows the heat-energy output and related parameters for the blanket for three-loop operation, based on the selected orifice arrangement. Table 4–6 lists similar information for the seed.

During the initial runs of PWR, it has been established that the core will be able to override peak xenon for an appreciable portion of its life.

During the overriding of such xenon, the amount of power will have to be reduced somewhat because of the more unfavorable flux patterns encountered. While analyses are still in progress, it appears safe to say that the power capability when overriding peak xenon early in life will be at least 75% of design rated power at equilibrium xenon with three loops operating. Thus xenon presents no problem in handling normal load transients.

4–4. Fuel-Handling and Refueling Equipment

After the plant has been in operation, the fuel and reactor internal components are highly radioactive and cannot be handled by conventional means. Underwater fuel-handling facilities and equipment have been provided for operations involving handling radioactive pieces. The water provides the shielding necessary to reduce the radiation to a tolerable level at the water surface and also serves as coolant for any stored irradiated fuel.

As described in Chapter 1, equipment and facilities must be provided to permit refueling by any one of three methods. Equipment for this purpose is described below. Also discussed are the underwater and dry pit storage facilities provided.

As also described in Chapter 1, equipment is now being designed and procured for an alternate method of refueling. This differs from the method described in this chapter in that shielded coffins rather than water will be used to handle radioactive components. The shielded-coffin method is referred to as dry refueling; the water-shielded method is known as wet refueling. Actual use will determine the relative merits of the two schemes.

4–4.1 Core installation and removal. The core hoisting equipment shown in Fig. 4–24 is used to lift the core assembly and place it in the reactor vessel, or to remove it from the vessel. This equipment, designed so that all these operations can be performed remotely under water after parts have become radioactive, was also useful for initial installation of reactor components. Essentially, the equipment comprises a large grapple which is used to engage lifting bails on the core cage. Locking mechanisms prevent the grapples from dropping the load and an extension bar is provided to keep the crane hook and block above water when the reactor pit is flooded. The bar also prevents the possibility of lifting the core to an elevation that would permit radiation hazardous to operating personnel. A long drive shaft actuated by an operator above the canal locks and unlocks the grapple from the core. An alignment jig, also shown in Fig. 4–24, is placed on top of the reactor when the reactor core is installed or removed. This jig is a steel framework with vertical guides and

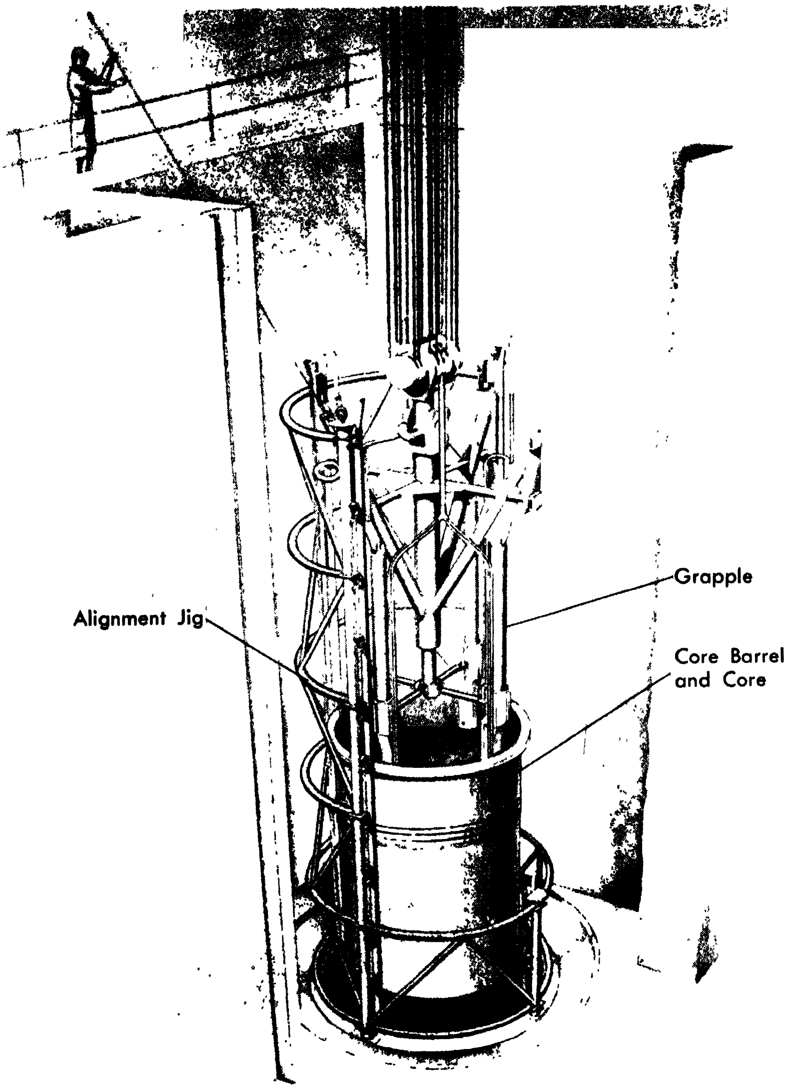


FIG. 4–24. Core installation and removal.

keys on the bottom flange for alignment and positioning on the reactor vessel. Rollers on the core hoisting grapple engage in guides on the jig and align the core as it travels into or out of the vessel. The jig also aligns the grapple hooks to engage the core bails before the core is removed from the reactor.

The core hoisting equipment is also used to handle the thermal shields, the hold-down barrel, and the core support spring assembly.

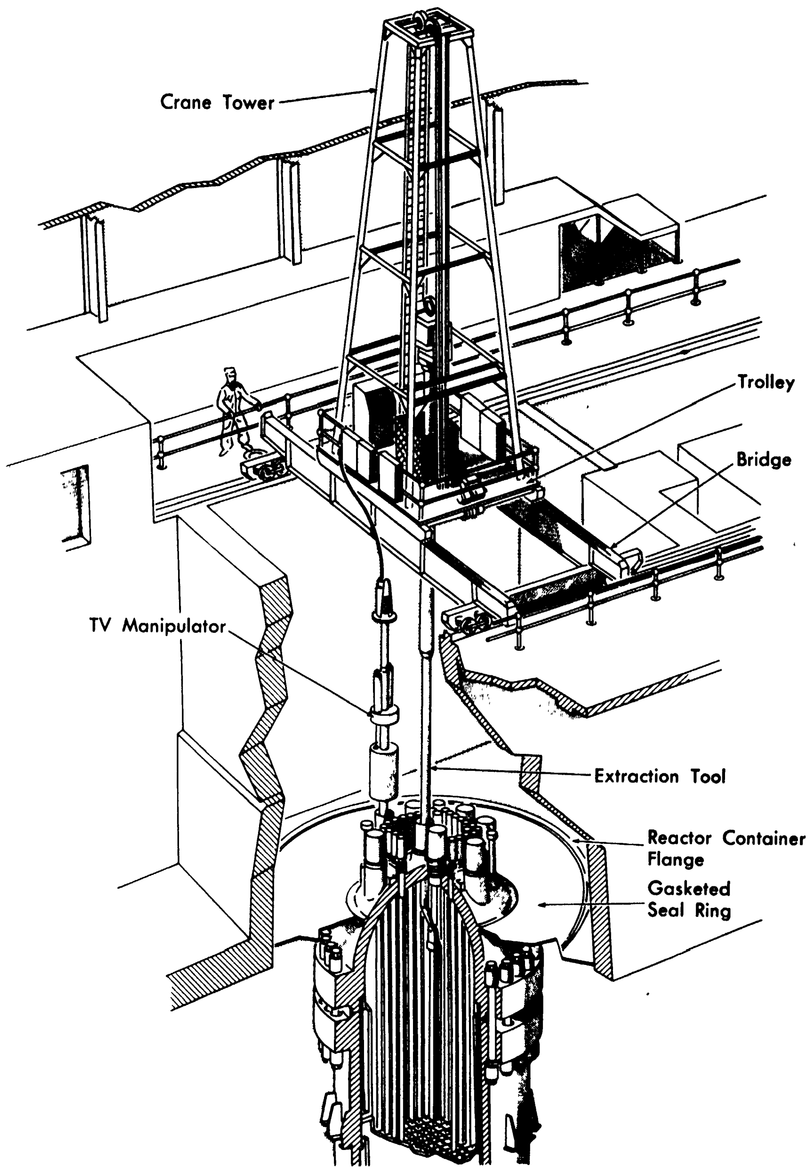


FIG. 4-25. Extraction tool in operation.

4-4.2 Reactor refueling. The fuel assembly extraction tool, which handles individual fuel assemblies, is a long grappling device 9 in. in diameter over the portion that enters the reactor vessel through any of the ten access ports in its head. Figure 4-25 shows the tool in operation. The lower end of the tool consists of a grappling head and an extension

mechanism to align the grappling head with assemblies. The upper end of the tool, which is always above water, contains the drives for the tool head motion. A hoisting device mounted on a trolley above the vessel moves the tool vertically, as necessary, during the fuel-handling operations.

In Fig. 4–25, the tool is shown operating through a central access port and engaged to a blanket assembly being lowered for insertion into the core. The figure also shows the diaphragm seal that is in place during through-the-head refueling. By closing the gap between the bottom of the reactor pit and the reactor vessel head, this seal permits underwater refueling without flooding the plant container space around the vessel.

4–4.3 Fuel services. As indicated in Chapter 16, the fuel-handling building houses the reactor container and contains facilities for installing and removing fuel assemblies. The sections of the building directly concerned with fuel handling are shown in Fig. 4–26. A clean room at one end of the fuel-handling area contains an assembly stand and crane (Fig. 4–27). A door in the ceiling of this room permits passage of a core

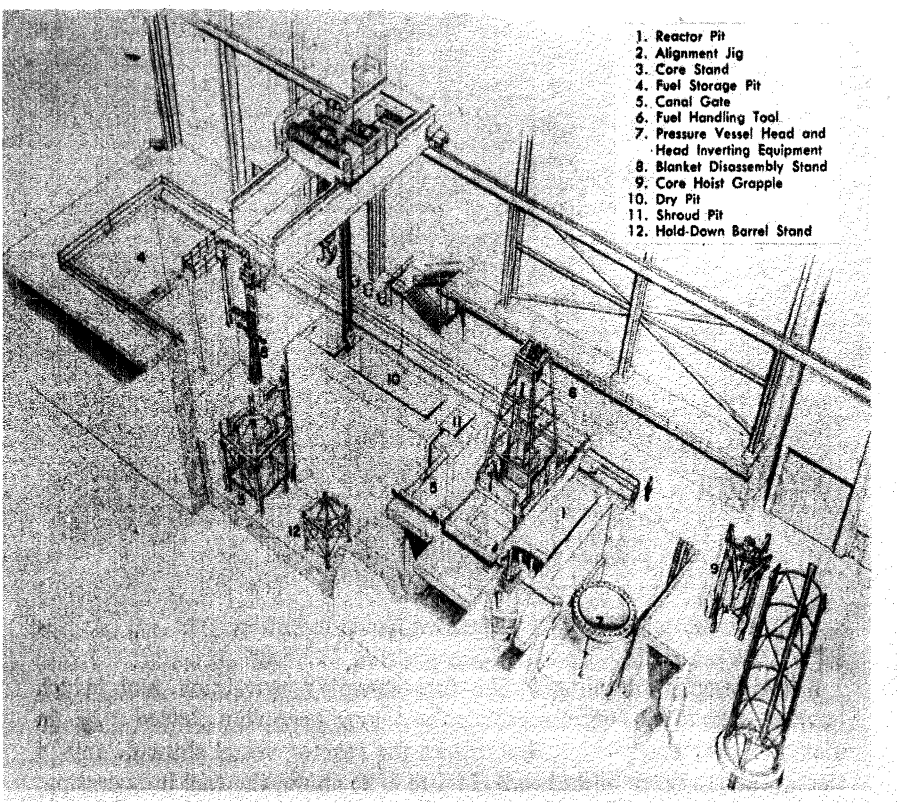


FIG. 4–26. Fuel-handling facility.

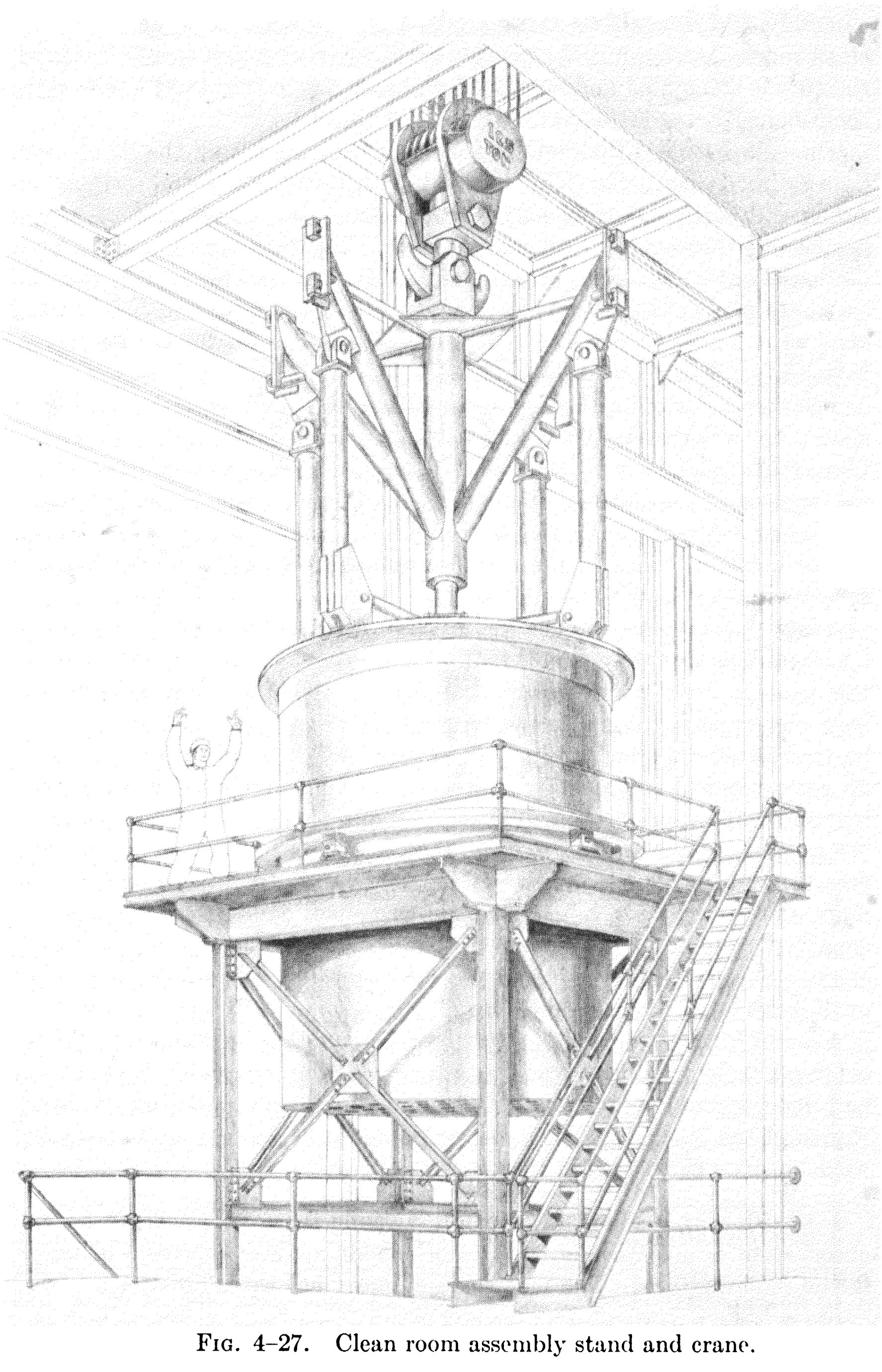


FIG. 4–27. Clean room assembly stand and crane.

assembly and also allows the main fuel area service crane access to the clean room. During initial assembly and whenever a new core is installed, the core is assembled and checked for alignment on the stand in the clean room before being installed in the reactor vessel.

The compartment fuel-handling canal is separated from the clean room by a railroad dock. Flooding the canal with water establishes a shield between radioactive components removed from the vessel and personnel performing disassembly and maintenance. The fuel storage area is used for underwater storage of irradiated fuel assemblies obtained in the disassembly of a core. Removal of the transfer canal gate and fuel storage area gate makes it possible to transfer the fuel assemblies underwater from the reactor pit to the fuel storage area.

The core disassembly area is separated from the fuel storage area by a gate. An underwater stand, capable of accommodating two core assemblies, is installed in this area. When the core assembly is removed from the reactor vessel during refueling, it is placed on this stand for disassembly or replacement of fuel assemblies. As discussed later, additional equipment is provided in the canal to remove and replace or redistribute bundles in the blanket assemblies.

A lock between the core disassembly area and the dry pit serves to transfer nonradioactive components from the canal to the dry pit. The gate to the canal is removed to permit components from the canal to enter the lock. By replacing the gate and pumping the water out, components may be transferred to the dry pit. The dry pit provides an area for working on components that do not require shielding and for maintenance, testing, and calibration of the extraction tool and the crane that carries it.

Components are transferred from the reactor pit to the core disassembly area through the canal. A removal gate that separates the canal from the reactor pit permits draining the reactor pit without loss of water in the canal. A channel at the side of the canal connects the shroud pit to the canal. Shrouds removed from the reactor vessel are stored in the pit until reassembly in the reactor vessel.

A stand in the storage space for the reactor vessel head, next to the reactor pit, holds the head when it is removed from the vessel. The stand and dry pit provide facilities for cleaning and decontaminating the head. Repair of the closure seal weld, access port seal welds, and mechanism seal welds may be performed in the pit.

4-4.4 Viewing equipment. Remote viewing is necessary where normal observation is not feasible because of shielding water, extreme distance, or the radiation field. The viewing equipment has three functions:

(1) It permits observations inside the pressure vessel (through access ports) during refueling.

(2) It provides a means of observing operations in the canal.

(3) It makes possible detailed inspection of radioactive materials in color and with high resolution and magnification.

Both a closed-circuit television system and an optical viewing system are available. Television is used inside the reactor and over a core in the underwater assembly stand because it offers several advantages which optical methods lack: mobility, small size, erect images, and reduced exposure of personnel to radiation. Optical methods are preferred for detailed examinations because of their better image quality, notably in color, resolution, and depth.

When fuel assemblies are exchanged through the reactor vessel head, a television camera is inserted into the vessel through an access port near the one used by the extraction tool (see Fig. 4–25). Observation of the extraction tool by the television system is used only to supplement the instrumentation built into the tool, and to diagnose any unexpected situations that may occur. The tool can be used to perform all its normal operations without the television system.

The camera is encased in a watertight, water-cooled housing mounted on a boom that provides lighting and makes possible the manipulations necessary to see all the reactor vessel interior. In use, the boom is supported on the head of the vessel and is self-contained except for power and signal cables to the handling crane. Therefore, it does not interfere with other equipment operating over the reactor pit. The television boom can also be used in other portions of the canal by providing temporary support equivalent to that provided by the pressure vessel head.

The periscope shown in Fig. 4–28 is a commercial instrument mounted on the side of the fuel-handling canal. Objects to be examined are placed in front of its objective lens by the underwater handling equipment. The main crane with the core hoisting rigging is used to manipulate the core and other large pieces. The diameter of the periscope is small enough that it can be used for viewing inside the vessel, if necessary.

In addition to the specialized viewing equipment, two small viewers will be provided for casual or over-all observation of underwater operations. These glass-bottomed tubes can be inserted a few inches into the water, to minimize surface effects. They are equipped with removable binoculars for magnified images, if required.

Underwater lights placed at intervals around the canal provide general illumination. These lamps and their sockets are commercial equipment of a type used by divers. They are mounted on portable masts, as shown in Fig. 4–28, so that they may be grouped for brighter illumination at a particular spot and so that they may be removed from the canal for servicing.

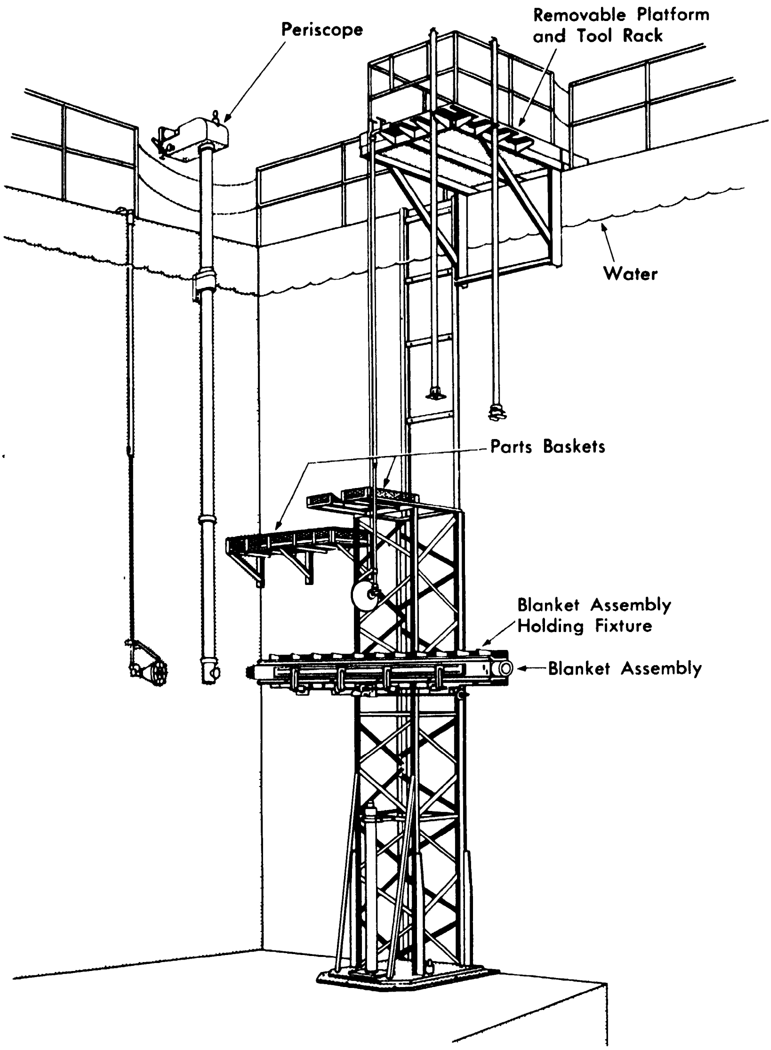


FIG. 4–28. Blanket disassembly stand and periscope.

4–4.5 Fuel maintenance equipment. A container (Fig. 4–29), is provided for shipping a single irradiated assembly to a laboratory for examination. The inner cavity is a stainless steel casing large enough to accommodate a 7-in.-square, 110-in.-long fuel assembly. The fuel assembly is immersed in water and surrounded by a stainless steel cooling coil capable of removing 40,000 Btu per hour. This heat removal rate will maintain the bulk water temperature at 160°F around a seed assembly that has been oper-

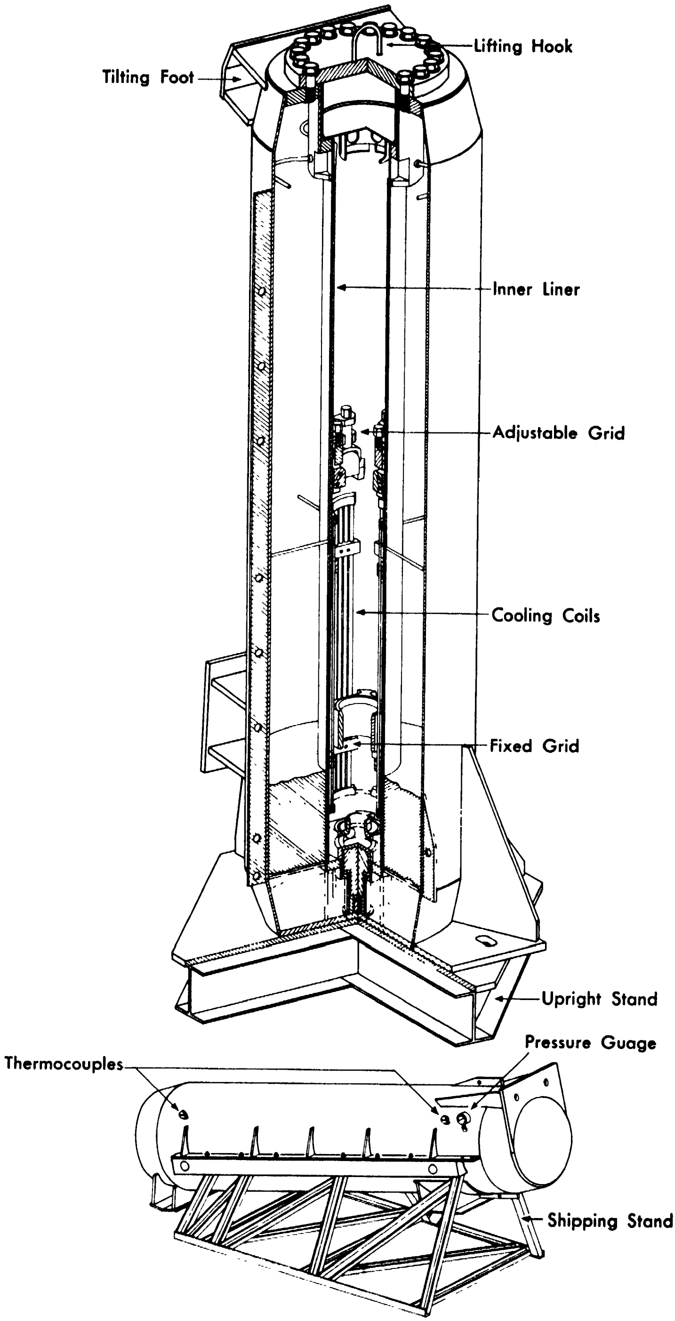


FIG. 4-29. Irradiated fuel shipping container.

ated for 600 continuous full-power hours and has been allowed to cool for 50 hr. The inner cavity is surrounded by lead shielding which is, in turn, enclosed in a carbon steel outer jacket. The outer jacket has a framework that supports the container at a slight angle from the horizontal so that cooling water will circulate through the center of the fuel assembly by convection. The framework will allow vertical loading of the container in the canal at Shippingport and horizontal unloading into the hot cells of the laboratory, where no canal is available. Standard casks will be provided for shipping spent fuel assemblies to processing points.

The fuel storage pit contains structural racks for storing a complete core complement of fuel assemblies, plus four extra seed assemblies and seven extra blanket assemblies. These racks are arranged in a rectangular pattern, the spacing for the seed assemblies being sufficient to maintain a subcritical array at all times even without control rods or poisons dissolved in the canal. There will, however, be a control rod, or shipping rod, in every seed assembly during storage. The racks are fabricated from stainless steel to resist corrosion, since they are not accessible for maintenance.

Blanket disassembly equipment, shown in Fig. 4–28, is provided in the canal for remotely dismantling and assembling core blanket assemblies. This equipment permits changing the orifice assemblies and replacing the bundles with new bundles, reusing the Zircaloy shell and other components. Included in the disassembly equipment are all the tools necessary to dismantle the blanket assembly remotely, baskets for handling bundles and parts, a tool for transporting the blanket assembly through the canal, a working platform, and a removable stand designed to grasp and hold the blanket assembly and also to rotate it for viewing the ends with the underwater periscope.

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CHAPTER 5

FUEL ELEMENT DEVELOPMENT

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5–1. Introduction

The Shippingport seed-blanket core arrangement necessitated the use of two distinctly different types of fuel element—the seed of highly enriched uranium and the blanket of natural uranium. Since elements of the former type had been extensively investigated previously, a very major portion of the fuel element developmental and testing effort was directed toward the blanket element and materials. The development of both seed and blanket elements is described in this chapter but, in keeping with the relative amounts of effort, the discussion is devoted primarily to the investigation, development, and testing of blanket elements.

5–1.1. Factors influencing fuel material and fuel element development. The types of fuel materials under consideration for Shippingport were sharply limited by the criteria adopted for fuel elements and fuel material. These criteria are listed in approximately the order in which they influenced fuel material development: (1) It was deemed essential that the fuel element possess the highest possible uranium loading with a minimum of additional neutron-absorbing constituents. This requirement evolved from the nuclear design of the core. (2) The fuel should possess sufficient radiation stability, even at high burnup levels, so that the amount of dimensional instability or volume change of the fuel would not result in cladding rupture. (3) The fuel, even at high burnup levels, should possess sufficient corrosion resistance to high temperature water so that even if a number of elements did fail, reactor operation would not be impeded or over-all core life significantly reduced. Even with a radiation-stable fuel, initial defects in the cladding or closure welds might cause failure during operation. In addition, the combined effects of irradiation and corrosion on an initially defective fuel element should not cause a progression of failures to adjacent fuel elements by restricting or blocking cooling channels.

Several other considerations also guided the program. One of these was that the fuel element and fuel material should ultimately be adaptable to low-cost quantity production; another, that the fuel material selected should be promising for future improvement. All these considerations

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led to intensive metallurgical research and development, because no fuel materials were known which met all or even a significant fraction of the requirements.

5–1.2 Historical development of metal and oxide fuels for the blanket. When the Shippingport project was instituted, development and operational experience had been accumulated on a limited number of types of fuel materials and fuel elements in high power density reactors. Such fuel materials were in the form of plates of zirconium and of aluminum base alloys containing relatively small amounts of enriched uranium or of relatively massive slugs of commercially pure uranium.

The plate form fuel materials either were chemically highly corrosion resistant to high temperature water because of the uranium dilution, or were applied in low temperature water reactors, such as the Materials Testing Reactor (MTR), in which they possess a high degree of corrosion resistance. Experience with the metallic uranium fuel slugs exposed in relatively low temperature water indicated that a certain percentage of fuel elements were likely to fail. Extrapolating this experience to anticipated PWR burnups and lives indicated that, even with greatly improved integrity, fuel element failures could be expected to cause an unacceptably large number of reactor shutdowns.

The developmental program was first centered on improving uranium by alloying to make it corrosion resistant to high temperature water. The goal of this program was to develop a fuel, containing at least 75 vol. % uranium, able to withstand corrosive action of water or steam at temperatures up to 750°F through cladding defects for at least one year under reactor irradiation. Also, the fuel was to be stable with respect to dimensional or volume changes at center temperatures up to about 850°F for burnups of at least 5000 Mwd/t. These requirements demanded the inception of a large-scale research program on alloy development, physical metallurgy, and corrosion. Exhaustive alloy development investigations disclosed only two types of materials exhibiting the necessary qualities to a sufficient degree. These materials were (1) alloys in which the gamma or body-centered cubic phase of uranium was stabilized at reactor operating conditions by additions of molybdenum (in amounts up to 12 w/o), niobium, or combinations of both, and (2) alloys of uranium containing 3.8% silicon corresponding to the composition of the intermetallic compound U_3Si. Further investigation eliminated the U_3Si alloy from consideration because a practicable fabrication process was not found for cladding the material and thus retaining its corrosion resistance and radiation stability.

To carry out the above alloy development program, it was necessary to devise, build, and operate in-pile loop equipment in which high temperature

water could be circulated in contact with exposed fissionable material. Such equipment had never previously been built.

The characteristics of corrosion failure of improperly fabricated fuel elements containing gamma-phase uranium-molybdenum alloys were such that disastrous splitting would occur in the cladding, and finely divided uranium oxide would be released into the reactor system. The decontamination that would be necessary if this release occurred could be performed only with considerable difficulty. Since it was not obvious that a fabrication process could be developed that would be sufficiently well controlled to ensure the absence of such effects, the applicability of this material to PWR began to be seriously questioned. Furthermore, the use of molybdenum was unattractive as an alloying addition because of its relatively high thermal neutron absorption characteristics. The discovery of a hitherto unsuspected high epithermal resonance absorption peak made the use of molybdenum even more unattractive.

Some work on uranium dioxide as a possible fuel material for this reactor had been started. A relaxation of one of the requirements originally imposed, namely that the fuel material contain in excess of 75 vol. % uranium, permitted consideration of using UO_2 in the blanket. Even with the lower uranium content, the reduced neutron poisoning effect of UO_2 made it more desirable than the alloy of uranium with 12% molybdenum. It was found that UO_2 satisfied the other requirements of corrosion resistance and irradiation stability. Furthermore, the ability of the cladding to retain UO_2 in bulk form, even in case of severely failed fuel elements, indicated that decontamination problems would be much less severe than with uranium alloy fuel rods. Thus UO_2 was selected as the reference fuel material for the blanket of the first core.

The resistance of UO_2 to dimensional and corrosion damage is surprising, considering that the fission process adds, in appreciable quantities, some 40 extraneous elements to the UO_2 lattice. For example, after a burnup of 10,000 Mwd/t, about 0.7% of the atoms in the UO_2 lattice are fission products. In addition to its excellent dimensional and corrosion properties, UO_2 has been found to retain fission gases very well, which means that high gas pressure will not build up between the UO_2 fuel and the cladding of the fuel rods. It is postulated that this ability to retain fission gases results from their accommodation in the UO_2 lattice.

In-pile tests at various heat fluxes have shown that UO_2 is a satisfactory material for use in the PWR core. In fact, early in the development stage, fuel rod samples that were actually touching each other (rather than having the 0.055 in. clearance) did not fail when tested in the MTR. In addition to proving the performance of a single rod, this test showed that a failure of one rod would probably not cause failure of its neighbor. Another confirmation of the usefulness of the PWR fuel element was obtained by

operating both sound and intentionally defected fuel-element samples in a high-temperature water loop in the Chalk River NRX-reactor for 14 months. There was no indication of failure after more than 10,000 Mwd/t burnup at maximum PWR Core I heat fluxes.

Considerations of susceptibility to progressive damage failure, ability to fabricate fuel elements in quantity at low price, and irradiation stability led to the choice of cylindrical fuel elements consisting of an outer tube of Zircaloy containing pellets of UO_2. The diameter of the fuel element was selected on the basis of anticipated maximum blanket heat fluxes and available thermal conductivity data to limit central UO_2 temperatures to those below the melting point of UO_2. As a result of system contamination considerations, a 9½-in. length of fuel material was chosen, so that failure of a fuel element would expose only about 150 g of UO_2 to the reactor coolant water.

5–2. Development of Fuel for the Blanket

5–2.1 Problems in use of UO_2 fuel. When UO_2 was chosen as the reference fuel material for the Core I blanket, the metallurgical development program was directed toward solving those problems that appeared most important in validating the material for power reactor use. During the course of the program, other problems were revealed which required solution; certain of the problems initially visualized as important proved to be unimportant with respect to the final application of UO_2 fuel.

One of the principal items of concern was the determination of pertinent physical properties of UO_2. Possibly the most important of these is thermal conductivity, since this property to a large extent fixes allowable fuel element dimensions and permissible heat-transfer rates. The melting point is also important in this connection, since if the permissible central temperature is considered to be limited by the phase change occurring at the melting point, this property also determines fuel element dimensions. Disagreement in values of the melting point of as much as 400°C was reported in the literature. The strength properties of the sintered compacts were considered important, since they control the thermal fracture characteristics of the fuel pellets. Properties such as thermal expansivity and specific heat were considered to be already sufficiently well known.

The chemical behavior of UO_2 was obviously important. It was necessary to establish that UO_2 either was inert in water or could be rendered inert by suitable changes in water composition. Some preliminary indications revealed that UO_2 was subject to attack during high-temperature testing in water under certain conditions and, furthermore, that certain types of UO_2 powder, particularly those prepared from uranium hexafluoride, caused corrosive attack on the Zircaloy-2 cladding. Earlier work done at Bettis had indicated that zirconium was highly reactive with oxide crucible

materials during melting and that it reacted with many ceramic materials during heat-treatment operations at submelting temperatures. It was considered necessary, therefore, to measure the temperature and times at which the rate of reaction between the Zircaloy tubes and the UO_2 fuel pellets became appreciable and to determine whether such conditions were likely to be encountered during fuel element fabrication or operation.

Since UO_2 is produced as a fine powder, it can react with air at ordinary temperatures to form higher oxides of uranium. The effects of such higher oxidation states upon the subsequent behavior of the powder were unknown, so considerable work was done on the storage behavior of UO_2 powders and on the rates of their reaction with air. Hydrogen-reduced UO_2 produced by the Mallinckrodt Chemical Works (MCW) was chosen as the reference material from which the blanket was to be fabricated. It was considered undesirable to purchase such material without complete knowledge of such pertinent characteristics as particle size, porosity distribution, surface area, etc., since undetected changes in the reduction process could seriously affect the finished product. Therefore, a program of powder characterization was instituted—not only for powder prepared by the Mallinckrodt process, but also for that produced by other suitable methods.

It soon became evident during the development of the UO_2 fuel that a highly sintered product was desirable for Shippingport application. Therefore, the effects of powder properties and of pressing and sintering conditions on the sintered compact's open and closed porosity, density, grain size, composition, and metallographic structure were investigated.

The nature of the UO_2-oxygen phase diagram, at least within the ranges of oxygen content with interest for Shippingport, was also important. The nature of the binding of the uranium and oxygen ions in uranium-oxygen compounds and the mobility of each ion were visualized as important quantities in evaluating the behavior of the material.

Of special concern was the behavior of UO_2 during reactor radiation, particularly in regard to the following:

(1) Heat-transfer characteristics of fuel elements operating both undefected and defected, and measurements of the heat fluxes at which center melting occurred, were considered basic.

(2) The high thermal expansivity, low thermal conductivity, and poor strength of UO_2 indicated that it might fracture under the thermal gradients attained during fuel element operation. The extent of such fracture, and the determination of the conditions under which possible progressive fracture might cause the cladding to deform or fail as a result of a "ratcheting" process, were tested both in- and out-of-pile.

(3) Similarly, it was visualized that during reactor down periods water might enter an unbonded UO_2 fuel element through a small cladding

defect and subsequently flash to steam when the reactor was brought to power; this might develop excessive internal pressures and lead to gross failure of the cladding with consequent blocking of water channels. This "waterlogging" type of failure was extensively investigated.

(4) Since one of the criteria for successful operation of the fuel elements is that failure of an individual fuel element must not cause adjacent elements to fail, it was necessary to establish experimentally that such progressive failures would not be an important factor in reactor core operation.

(5) Since the UO_2 was not bonded to the Zircaloy cladding, the release of fission products into the coolant from a defect in the cladding would be much greater than in the case of a bonded corrosion-resistant metallic fuel. The nature and amounts of the fission products released and their effects upon coolant contamination were prime concerns. Similarly, volatile fission-product release, particularly that of the noncondensable gases xenon and krypton, could cause buildup of internal pressures and temperatures severe enough to require measurements of these quantities.

(6) Effects of irradiation upon both the crystal structure and the metallographic structure of UO_2 were also considered to have a vital bearing upon the performance of this fuel material.

To obtain definitive answers to the questions raised by the above considerations, a broad-based program was instituted at the Bettis Plant and was supplemented by subcontracts at installations such as Battelle Memorial Institute, Armour Research Foundation, Westinghouse Research Laboratory, Carnegie Institute of Technology, and Corning Glass Works. To coordinate these efforts and to promote rapid interchange of information among organizations performing research and development on oxide fuel materials, a UO_2 panel, which met frequently, was instituted. This panel consisted of representatives of Bettis Plant, the Atomic Energy Commission, Oak Ridge National Laboratory, Battelle Memorial Institute, Armour Research Foundation, Corning Glass Works, Hanford Engineering Works, Mallinckrodt Chemical Works, National Bureau of Standards, Massachusetts Institute of Technology, and Knolls Atomic Power Laboratory. In-pile loop irradiation tests were performed in the NRX reactor in cooperation with Chalk River personnel as part of a joint AECL-USAEC program. It was hoped that, in the less than $3\frac{1}{2}$ years between the inception of work on UO_2 and the beginning of operation of the Shippingport Station, sufficient information could be accumulated to permit selection of proper fabrication conditions, suitable fuel element dimensions, and allowable operating conditions. These conditions could assure, to as high a degree of certainty as possible, successful operation of the fuel elements in the plant.

A fuller description of the results of this program is contained in the two reports WAPD-183, "Effects of Irradiation on Bulk UO_2," and

WAPD-184, "Properties of UO_2," both of which are available from the Office of Technical Services, Washington 25, D. C.

5-2.2 Results of developmental program. *Out-of-pile evaluation.* The initial design of the Shippingport fuel element was based upon thermal conductivity data derived by Kingery et al.* These data indicated that at reference heat fluxes, center temperatures would be attained somewhat below the sintering temperatures of 1700 to 1750°C. However, new thermal conductivity measurements yielded values about 25% lower than those previously reported. These new data elevated calculated center temperatures to the neighborhood of the melting point of UO_2 (additional work has indicated that the melting point of UO_2 is about 2750°C). Data obtained are plotted in Fig. 5-1. Although the absolute agreement of the two sets of data is quite good considering the different types of UO_2 powder used in preparing the samples and the quite different measurement

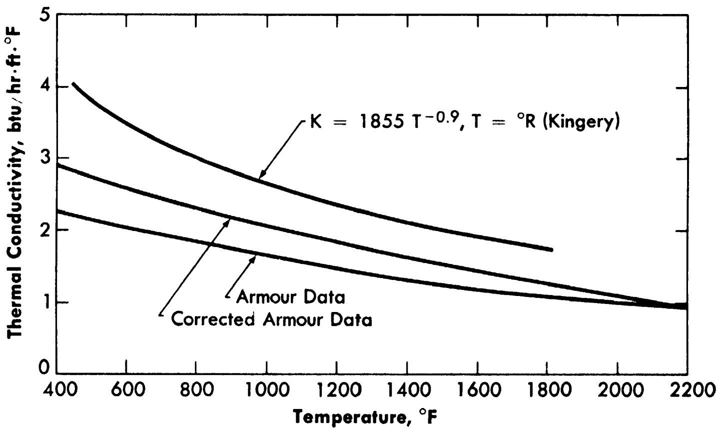


FIG. 5-1. Armour Research Foundation data on thermal conductivity k of UO_2 vs. temperature, measured at 75% of theoretical density and corrected to 95%, together with empirical curve of Kingery et al. for 95% density.

techniques employed, the approximately 25% difference in the values noted is highly significant with respect to the calculated center temperatures. Significantly lower thermal conductivity values are exhibited by nonstoichiometric (of composition $UO_{2.1}$) than by stoichiometric $UO_{2.0}$. This may serve to explain the reduced thermal output capabilities of fuel

*W. D. Kingery, J. France, R. L. Cable, and T. Vasilos, "Thermal Conductivity: X, Data for Several Pure Oxide Materials Corrected to Zero Porosity," *Journal of the American Ceramic Society.* Part II, Vol. 37, No. 2, pp. 107-110 (1954).

rods operated with defects in the cladding; such defects allow access of steam to the fuel, with resultant oxidation of the stoichiometric $UO_{2.0}$ contained in undefected fuel rods.

Modulus-of-rupture data, obtained over a range of temperatures up to 1000°C, indicated that the UO_2 might fracture under the stresses imposed by the thermal gradients encountered during operation. This expectation was confirmed not only by irradiation tests but also by out-of-pile cycling tests of sample fuel elements that were rapidly cycled between temperatures of 600 and 1500°F. However, even after thousands of such thermal cycles, the cracking pattern observed after the first few cycles remained unchanged, consisting of only about four to five cracked pieces per original UO_2 pellet. No trace of powdering or ratcheting, such as would be induced by progressive disintegration of the pellets, was noted.

Thermodynamic calculations showed that UO_2 should react to form higher oxides in pure water containing dissolved oxygen, but that the UO_2 structure would be retained in water containing dissolved hydrogen. These calculations were confirmed by prolonged tests in high-temperature hydrogenated water; the tests revealed no chemical or physical changes after exposures of one year. Similar results were found with samples exposed to steam at temperatures up to the melting point of UO_2. In water containing an excess of oxygen, reaction occurred which resulted in the formation of nonadherent hydrated uranium oxides. A further check on the validity of the thermodynamic calculations was obtained by experiments in which U_3O_8 exposed to hydrogenated water was reduced to an oxide of the UO_2 structure. It was thus confirmed that under normal reactor operating conditions UO_2 is inert to the coolant, and that during short periods of operation with accidentally highly oxygenated water the rate of attack is sufficiently slow to constitute no operational hazard.

Similarly, the characteristics of the reaction of UO_2 with air or oxygen were measured not only to provide information on the storage stability of the UO_2 powders but also to provide insight into the kinetics of oxygen ion mobility in UO_2. It was found that such reactions occurred in two stages (Fig. 5–2). The first stage was explained on the basis of diffusion of oxygen interstitially in the UO_2 structure to a composition of approximately U_3O_7; the second stage was characterized by surface nucleation and growth of a second phase, U_3O_8.

The reaction of Zircaloy with UO_2 was measured over a series of temperatures. This reaction was accompanied by the diffusion of oxygen into the zirconium matrix, formation of a metallic high-uranium phase to a restricted extent, and the diffusion of uranium into the zirconium matrix. However, this reaction was measurable only at temperatures above 1100°F and appreciable in magnitude only at temperatures exceeding 1300°F; therefore such reactions do not constitute a limitation to either the fabrication or the operation of the Shippingport fuel element.

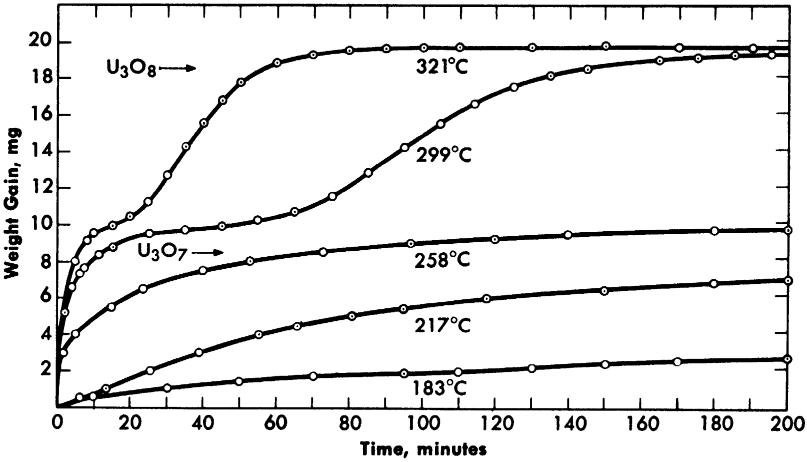


FIG. 5-2. Experimental rate curves for the oxidation of UO_2.

Processes were set up to characterize various types of powder with respect to particle size, surface area, porosity, density (both real and apparent), composition, etc. Not only was MCW powder (prepared by hydrogen reduction of UO_3) investigated, but also powders prepared by high- and low-pressure steam oxidation of uranium and uranium hydride, by precipitation and subsequent pyrolysis of ammonium diuranate, and by hydrogen reduction of both U_3O_8 and UO_3. These varying techniques produced a wide range of particle sizes and powder characteristics. Consistent with the variation in powder properties, the pressing and sintering behavior of each of these powders also varied markedly. It was found that constancy in properties of the reference MCW powder could be obtained by ball milling. Since the fabrication variability in MCW powder could be considerably reduced by the use of high pressing pressures, further powder treatments (such as ball milling) or other sources of powder were not required for Core I. The relations between porosity, density, and grain size of MCW pressed-powder pellets are shown in Fig. 5-3. As the course of densification proceeds (at constant sintering temperature and with increasing sintering times), crystal growth and the distribution of residual porosity between that communicating with the surface of the compact (open porosity) and that entirely contained in the compact (closed porosity) may be seen to follow a relatively complicated relation (Fig. 5-3).

These very small residual amounts of porosity were shown during irradiation experiments to have a profound effect on the rate of release of volatile fission products from the matrix. The electron micrographs of Fig. 5-4 show porosity changes associated with sintering. At low sintering densities (Fig. 5-4a) there are two classes of pores in UO_2 compacts: large irregularly shaped pores apparently located on the grain boundaries, and small regu-

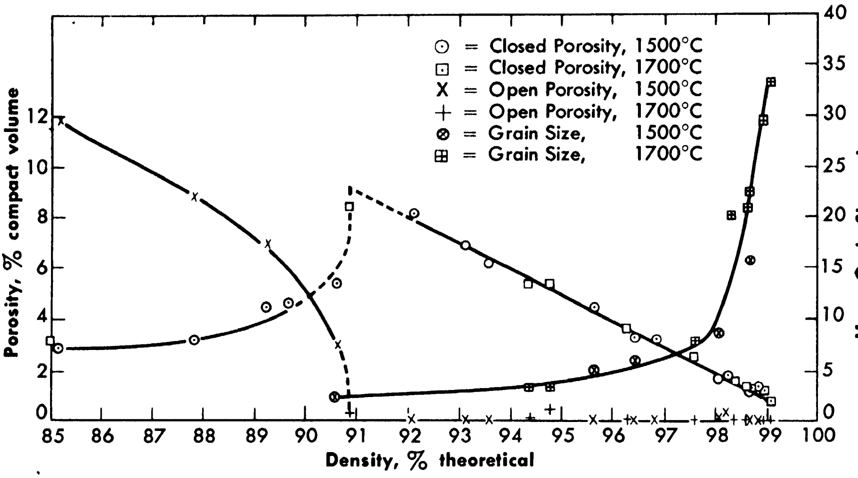
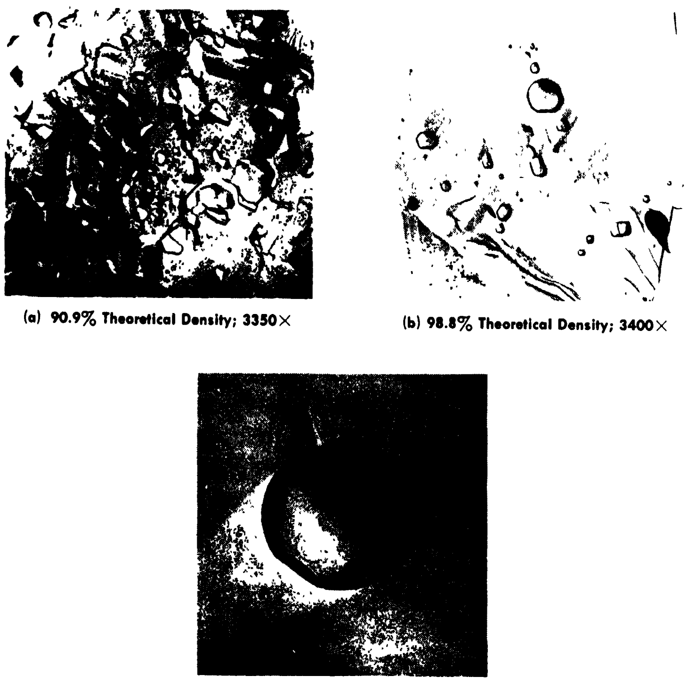


FIG. 5–3. Porosity and mean grain size of UO_2 pellets as a function of density. Pellets formed from wet ball-milled MCW UO_2 pressed to 65% theoretical density and sintered in hydrogen.



(a) 90.9% Theoretical Density; 3350×

(b) 98.8% Theoretical Density; 3400×

(c) 98.7% Theoretical Density; 16,000×

FIG. 5–4. Electron micrographs, made by the chromium-backed carbon replica technique, of pores in as-fractured ball-milled MCW sintered UO_2 compacts.

larly shaped pores usually observed in clusters inside the grains. As the UO_2 compact approaches theoretical density (Fig. 5-4b), the large pores develop regular symmetrical shapes and planes related to the crystallographic orientation of the grain in which they are located. At least some of the larger pores in high density UO_2 compacts (Fig. 5-4c) are bounded both by spherical surfaces containing concentric growth steps in the [110] crystallographic direction and by (100) and (111) planes in the symmetrical arrangement of a truncated octahedron. It has been found that the sintering of UO_2 is strongly affected not only by initial powder characteristics but also by the composition (excess O_2 content) of the compact or the nature of the sintering atmospheres. Thus the sintering temperatures required to attain densities in excess of 95% of theoretical may be reduced from 1700 to 1400°C by replacing the hydrogen atmosphere with a steam atmosphere, although the resulting compacts have a composition of $UO_{2.19}$ after such steam sintering.

Electromotive force measurements were performed on UO_2-O_2 solid solutions (by means of a solid electrolyte technique) to establish the thermodynamic equilibria data for this solution and thus to permit prediction of compositions attained during exposure of UO_2 to reactor environments. The self-diffusion rates of oxygen ions were measured by a O^{18}-exchange reaction in both stoichiometric and nonstoichiometric uranium dioxide in order to relate kinetics of oxidation and sintering with basic material quantities. Similar measurements of uranium ions established that they were relatively immobile in the UO_2 lattice.

In-pile evaluation. A major fraction of the developmental work at Bettis was related to the in-pile validation of the use of high density UO_2 as a fuel material. The several different types of tests performed were dictated primarily by the types of irradiation facilities available. One important class of tests was performed in high temperature water loops both in the Chalk River NRX reactor and in the Materials Testing Reactor. In such tests individual fuel element specimens fabricated with suitable variations of cladding-to-UO_2-pellet clearance, UO_2 pellet density, etc., were exposed at heat fluxes ranging from the average heat fluxes of the Core I blanket to values of 600,000 Btu/hr-ft^2, or almost double the heat flux experienced by the hottest fuel elements in the Core I blanket. In each experiment certain of the samples were exposed to the high temperature coolant through five mil diameter holes to simulate reactor operation under the conditions that failure of fuel element cladding would impose. The coolant contamination resulting from such defects is discussed in Chapter 7.

Another important class of tests was that in which clad samples were sealed in capsules in contact with a liquid metal alloy, NaK, as a heat transfer medium. These tests were performed, primarily in the MTR, to amplify the information on the fuel elements and their behavior obtained

from the in-pile loop tests. An important part of the information derived from such tests was the behavior of the fuel element during cycling of the samples in and out of reactor flux to simulate the effects of PWR startups, shutdowns, and power changes.

The final type of tests was proof testing of fuel element assemblies. A number of fuel elements joined in a tube sheet arrangement similar to that of the Core I blanket assembly were irradiated in high temperature water loops at both Chalk River and the MTR to provide information on the utility of the assembly design and its resistance to distortions imposed by unequal operation temperatures and hence varying thermal expansivities through the subassembly structure. A total of 360 fuel elements have either been irradiated and evaluated or are still undergoing irradiation in various reactor test facilities. Information has been obtained on the operation of such fuel elements at burnups as high as 25,000 Mwd/t, or double the peak burnup of Core I fuel elements.

One of the most important results of this testing and evaluation has been the establishment of the thermal behavior of fuel elements as affected by (1) operating heat flux, (2) the clearance between cladding and UO_2 pellets, and (3) the nature of the atmosphere in the annular clearance between cladding and fuel. With a helium atmosphere between cladding and fuel and with 0.001 to 0.004 in. diameter clearances, the reference clearances for Core I fuel elements, it was found that central temperatures in excess of 2750°C, as evidenced by melting of the UO_2 fuel, are not encountered at heat fluxes below 475,000 Btu/hr·ft^2. This is well in excess of Core I maximum heat fluxes. Because of the temperature drop through the heat transfer barrier afforded by the helium atmosphere or annulus, allowable heat fluxes decrease to about 400,000 Btu/hr·ft^2 as the clearance is raised to 0.005 to 0.008 in. In the case of fuel rods of small diametral clearance operating with a defect in the cladding, thus having the helium gas annulus replaced by a steam annulus of much poorer thermal conductivity and resulting in oxidation of the UO_2 to a higher oxidation state, the allowable heat fluxes before center melting occurs are reduced to about 400,000 Btu/hr-ft^2, or slightly greater than Core I maximum heat fluxes. Allowable heat fluxes do not appear to be sensitive to burnup, indicating that irradiation to at least 25,000 Mwd/t has but a minor effect on the thermal conductivity of UO_2. Similarly, diluting the helium atmosphere by release of fission gases such as xenon and krypton with much poorer thermal conductivity properties is indicated to have a minor effect on the fuel element thermal behavior. This arises from the restricted release of such gases, as discussed below.

Qualitatively, the thermal behavior of the fuel elements can be explained on the basis of thermal conductivity measurements of UO_2, cladding-to-pellet clearance, and the nature of the heat transfer medium between

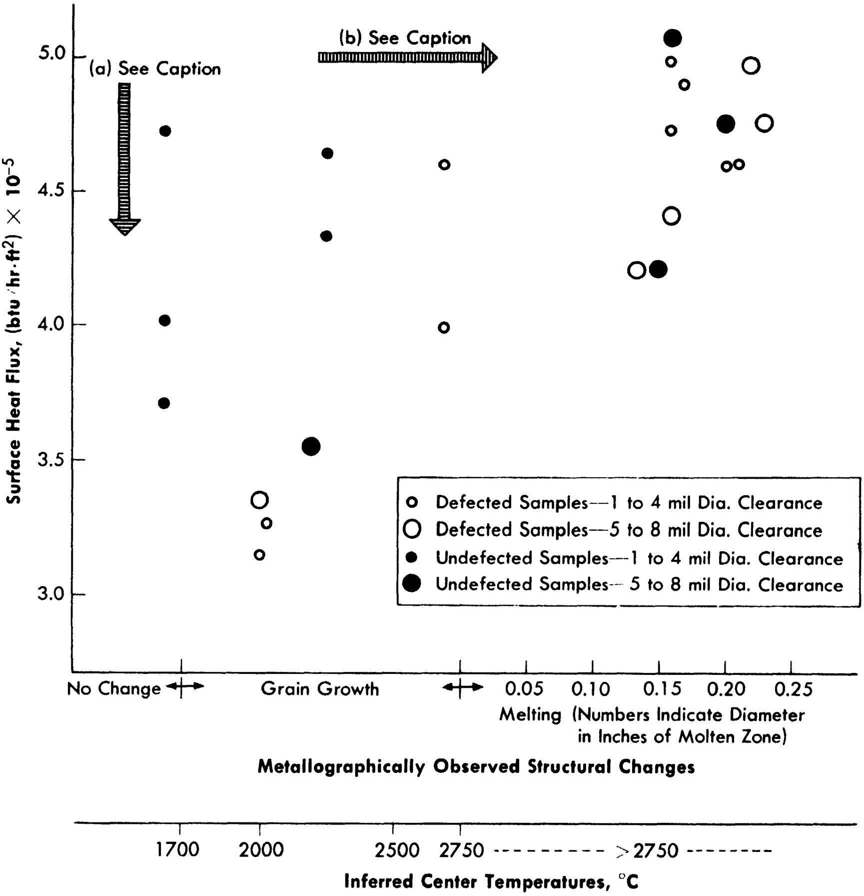


FIG. 5-5. Experimental observations of UO_2 melting diameters and grain growth (temperature in loop-operated PWR type fuel rods vs. surface heat fluxes). Samples of the same type (as indicated by open or closed points of the same size) yield (a) decreasing heat flux at constant temperature or (b) constant heat flux at increasing temperature because of an increase in clearance and/or defect between samples.

cladding and pellets; quantitatively, in many instances the onset of melting occurs at lower heat fluxes than would be anticipated. Thus, while empirical experimental observations have served to delineate the heat transfer capabilities of the Core I fuel element, the explanation of these observations on the basis of material properties still awaits experimental verification. The correlation between operating heat fluxes of various test elements and their observed thermal behavior with respect to metallographically observed melting is shown in Fig. 5-5. The effective thermal conductivity of UO_2 calculated from these observations is lower, by a factor of more

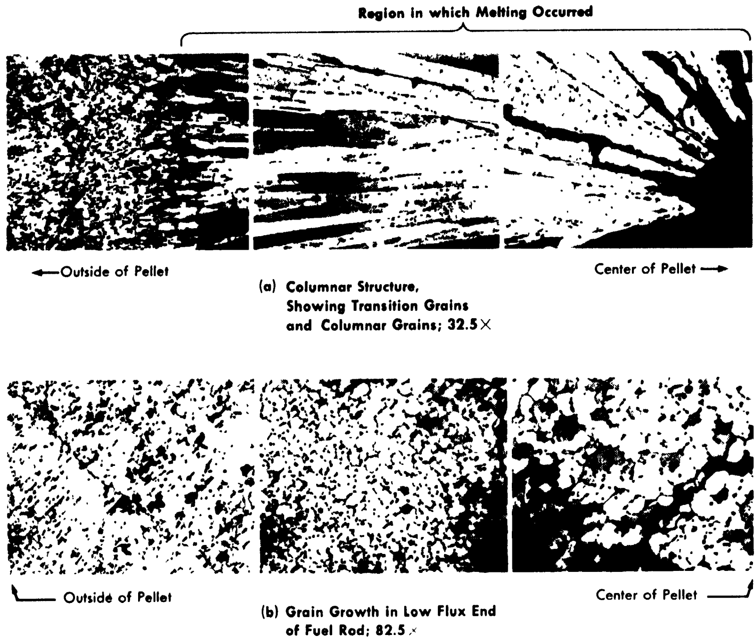


FIG. 5–6. Electron micrographs of UO_2 pellets, showing grain structure and grain growth (photo series, not composite).

than two, than the measurements reported for unirradiated UO_2. Figure 5–5 illustrates that either a defected sample (in which the helium is replaced by a steam atmosphere) or increased clearance between cladding and pellet reduces allowable heat fluxes. The type of melting and structural changes noted in fuel rods operated at excessive heat fluxes is shown in Fig. 5–6. Figure 5–6(a) illustrates the columnar appearance of those portions of the pellet diameter which have undergone melting and subsequent solidification and Fig. 5–6(b) illustrates grain growth in fuel rods operating with central temperatures above 1700°C but below 2750°C.

Proof tests of assemblies have shown that even in cases in which the heat fluxes between adjacent welded fuel rods have varied by as much as factors of 2, no permanent distortions of the subassembly have resulted. Experiments have been performed on subassemblies in which an attempt was made to simulate the most severe type of fuel element failure with respect to blockage of water channels, so that the probability of progression of failure from a severely failed element to an adjacent unfailed element could be assessed. In no case was it possible to observe any ill effect upon adjacent rods resulting from malfunctioning of a fuel element in a subassembly. The likelihood, therefore, of encountering progressive failure in Core I type fuel elements is considered highly remote.

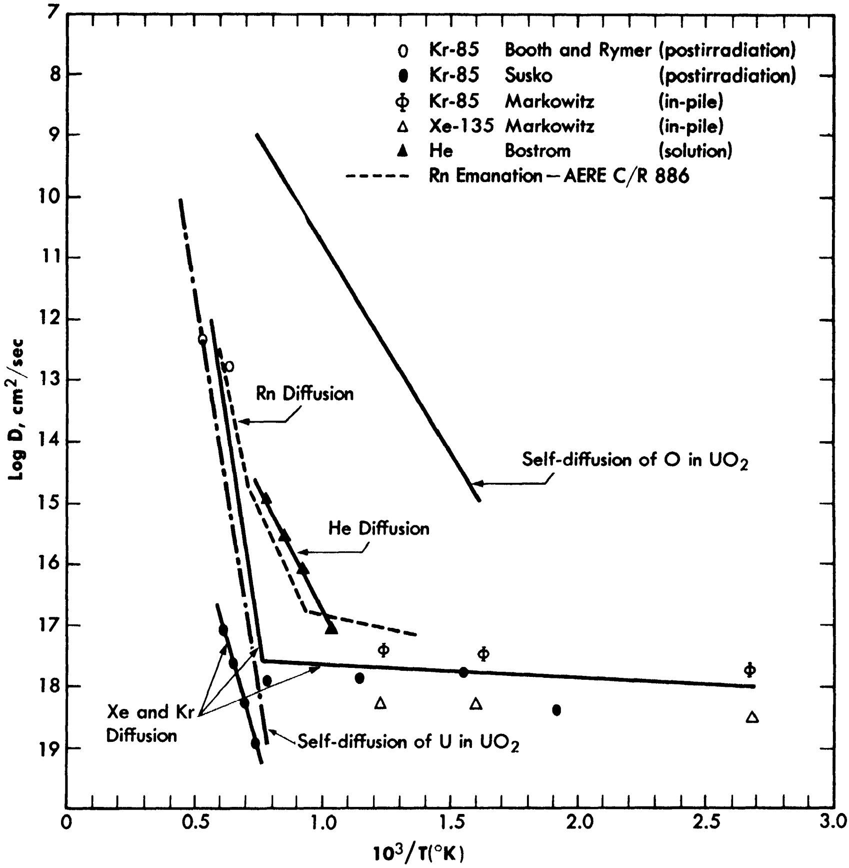


FIG. 5-7. Diffusion coefficients of noble gases in UO_2 as compared with those of oxygen and uranium ions.

It has been noted in using metallic fuels that as a result of formation of additional atoms upon fissioning and agglomeration of volatile fission products, particularly the noble gases xenon and krypton, within the metal structure, plastic straining of fuel element cladding and even rupturing have occurred. No trace of such changes has been noted in the case of UO_2, indicating that, at least for burnups of 25,000 Mwd/t, fission products can be accommodated within the UO_2 structure without marked structural changes. This behavior further adds to the utility of the Core I fuel element and its amenability to operation at extended burnups at elevated temperatures. The volatile fission product gases appear to be released from the UO_2 fuel material by a diffusion mechanism. The observed rates of the release of these fission products are consistent with a diffusion rate some six orders of magnitude less than that of oxygen ions in UO_2 and of the

TABLE 5–1

COMPARISON OF OBSERVED AND CALCULATED FRACTIONAL RELEASE OF Kr^{85} FROM IRRADIATED FUEL RODS

Specimen	Fuel diam., in.	Irradiation time, sec	Burn-up, Mwd/t	Density, % theor.	Radius of equiv. sphere, cm	Center temp., °C	Heat flux, Btu/hr·ft²	Fractional release: Observed	Fractional release: Calculated
X-1-G-3	0.357	4.85×10^6	2780	95	4.32×10^{-3}	2250	467,000	23×10^{-2}	32.3×10^{-2}
X-1-G-5	0.357	4.85×10^6	2060	95	4.32×10^{-3}	1700	404,000	16×10^{-2}	8.5×10^{-2}
X-1-B-4	0.242	3.63×10^6	1650	80	1.742×10^{-4}	990	147,000	9.5×10^{-2}	3.17×10^{-2}
X-1-C-6H	0.34444	5.61×10^6	350	95	4.32×10^{-3}	1500	367,000	5.8×10^{-3}	10.4×10^{-3}
WAPD-25-L-2	0.357	1.58×10^6	815	95	4.32×10^{-3}	1600	475,000	1.6×10^{-2}	4.7×10^{-2}
X-1-D-79-S	0.3444	8.67×10^6	5870	95	4.32×10^{-3}	1500	367,000	1.1×10^{-3}	40.8×10^{-3}
WAPD-29-1-1	0.357	2.42×10^6	860	93.7	3.16×10^{-3}	1000	230,000	3.0×10^{-3}	1.4×10^{-3}
WAPD-29-2-1	0.357	1.7×10^6	750	97.4	14.60×10^{-3}	1000	286,000	0.28×10^{-3}	0.26×10^{-3}

same order of magnitude as that of uranium ions. The observed diffusion rates of xenon and krypton are compared with those of oxygen, uranium, helium, and radon in Fig. 5-7. Since, by such a diffusion mechanism, fission gas release would be sensitive to temperature of operation, time of operation, and free surface exposed, it has been possible to relate the amount of fission gas release with characteristics of the initial compact as well as with operating conditions. The amount released increased markedly with higher operating temperature and longer periods of time available for diffusion to external surfaces. Measurement of surface areas of even highly densified sintered compacts has shown that real surface areas even in compacts sintered to 93% of theoretical density may be over 100 times greater than the geometrical surface area. Thus the observation has been confirmed experimentally that under similar exposure conditions, a fuel rod containing pellets of 97% theoretical density will release less than one-tenth the amount of fission gases released by a fuel rod containing 94% dense compacts. The correlation between amounts of fission gas release and those calculated on a basis of original compact density, operational temperatures and times, and the postulated diffusion mechanism of fission product mobility plotted in Fig. 5-7 is shown in Table 5-1.

The release of volatile fission products may have several consequences:

(1) As outlined in Chapter 7, such fission products contribute to contamination of the coolant and thus reduce plant accessibility.

(2) By admixture with the helium gas of the annular heat transfer medium, the conductivity properties of the fuel element can be diminished with a consequent detraction from the thermal performance capability of the fuel element.

(3) At sufficiently high burnup levels, sufficient pressure may be generated within the fuel element to rupture the cladding.

Both experiment and analysis indicate that such fission product effects do not limit fuel element performance for burnups of the order of five times the maximum PWR Core I burnups.

Cycling experiments, both in- and out-of-pile, have revealed that progressive fracture of fuel pellets does not occur during operation. Such fracture as is observed results primarily from imposed thermal stresses. Hence no degradation of compact structure is noted, and the degree of cracking is limited to that which can be reproduced in out-of-pile thermal shock tests. As a consequence, ratcheting of fuel and consequent deterioration of fuel integrity have not been observed. Another cladding failure mechanism postulated was waterlogging, in which water would penetrate the annulus within a fuel element and the pores of the UO_2 compacts during reactor shutdowns through defects in the cladding. Upon subsequent power operation, this water would expand into steam, creating excessive pressures within the fuel element cladding either from the inability of the

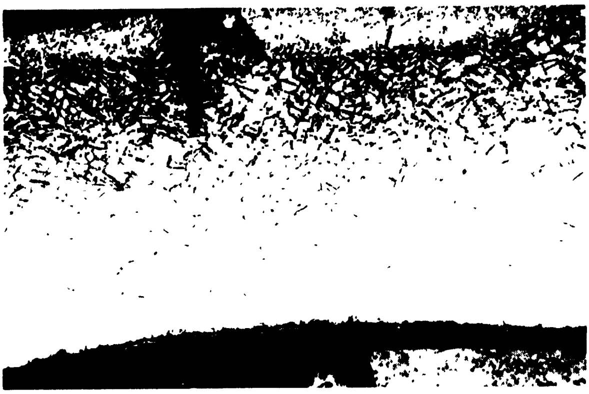


FIG. 5-8. Cross section of Zircaloy cladding showing hydride formation at outside of cladding (100×).

steam to escape through the small cladding defects or from clogging of such defects. While indications of such waterlogging behavior were encountered in the case of low density compacts or those containing unsintered powder, none of the tests with high density compacts have shown the existence of such a condition. It is concluded therefore that the occurrence of such waterlogging failures is highly unlikely if, as in PWR, high density sintered compacts are utilized.

Examination of the microstructure of compacts irradiated to burnups of up to 25,000 Mwd/t has revealed no microstructural changes other than those anticipated from the temperature of operation. Thus, grain growth is noted at operational temperatures in excess of 1700°C, the sintering temperature of the compacts, and evidence of melting at temperatures above the melting point (Fig. 5-6). However, no microscopic evidence of microstructural change, grain growth, pore formation or disappearance, or precipitation of second phases has been noted at lower temperatures of operation. X-ray examination of irradiated UO_2 has revealed only a slight line broadening; in contrast, irradiation of U_3O_8 powder, as shown by x-ray examination, resulted in almost complete disappearance of diffraction peaks. Thus, UO_2 appears capable of withstanding irradiation at high temperature with minimum structural change.

A number of failures have been noted in defected UO_2 fuel rods operated in high temperature water loops. These failures consisted of defects or cracks in the cladding and were at locations different from those of defects intentionally fabricated into the test rods. A high local concentration of hydrogen in the zirconium cladding was observed at each of these failures. The hydrogen migrated to the outside of the cladding by a thermal diffusion process. As a result of this high local concentration of hydrogen,

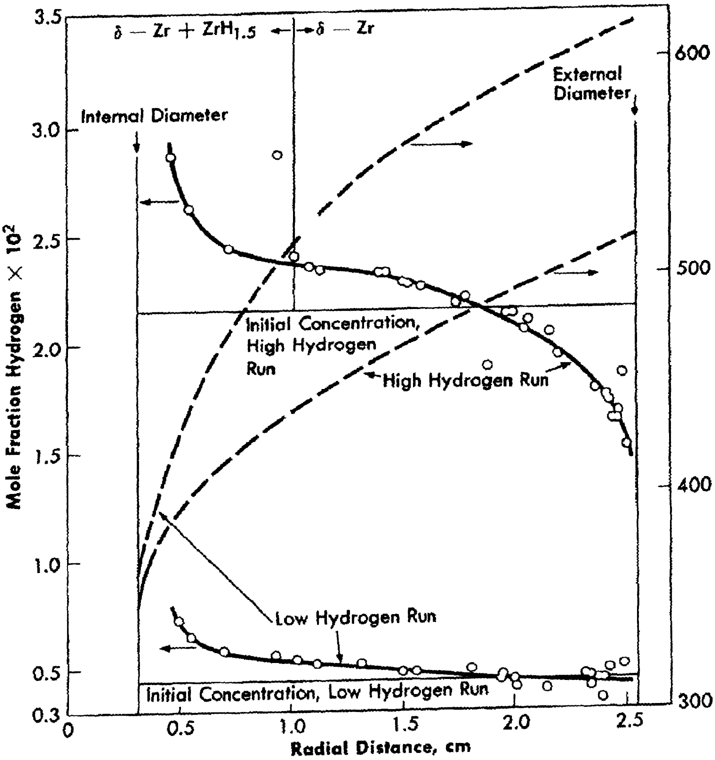


Fig. 5-9. Redistribution of hydrogen in thermal gradient of an internally cooled Zircaloy-2 cylinder.

there was a marked decrease in ductility and consequently an increase in susceptibility to cladding fracture from thermal stresses. The diffusion of hydrogen to the cold regions in the presence of a temperature gradient is illustrated in Fig. 5-8, which shows the hydride phase segregated at the relatively cooler surface of the cladding of a fuel rod. Failures of this type are typical for fuel rod samples fabricated with intentional defects and irradiated one to three months at high heat flux ratings in high temperature water. As shown in Fig. 5-9, out-of-pile experiments confirmed the segregation of hydrogen by thermal diffusion to the low temperature region of Zircaloy specimens held in a thermal gradient. Experiments show that sound undefected rods have not failed by this mechanism; sufficient hydrogen is not absorbed by undefected zirconium cladding during fabrication or operation to cause such failure.

The experimental program on Core I blanket fuel rods thus served to validate the use of an oxide fuel material for reactor use, establish the limits of its performance capabilities, determine suitable conditions for fuel

material fabrication and for fuel element design, and set a basis for interpretation of the performance of Core I blanket rods.

5.3 Development of Zirconium Fuel Alloy for Seed

As discussed in other chapters, the required seed lifetime dictated a U^{235} loading of 75 kg. The seed fuel alloy is nominally 0.04 in. thick; the Zircaloy-2 cladding, about 0.015 in. The resulting fuel alloy composition of 6.7 w/o uranium (93% U^{235}) in zirconium lies within the range of compositions that have been extensively investigated and developed. The principal problem in validating this material was to determine its resistance to failure under the high burnup conditions to which the seed fuel is subjected.

Information is presented below first on the unirradiated *structure* and *properties* and then on the *effects of irradiation* of the Zr-U fuel alloy. Also discussed are *zirconium, Zircaloy-2 cladding* material, and *hafnium* for control rods.

5-3.1 Structure of fuel alloy. Under the conditions of Shippingport operation, the maximum fuel alloy temperature has been calculated to be about 750°F. At such temperatures the fuel alloy is entirely in the α-zirconium range. Since the maximum solubility of uranium in zirconium at the eutectoid temperature of 610°C is of the order of 0.75 w/o, most of the uranium exists as a compound. In the binary zirconium-uranium system, this compound has the nominal composition of Zr_2U and is an ordered form of the body-centered cubic zirconium-uranium solid solution stable at temperatures above 610°C. The ordering reaction in this compound has been found to be quite rapid, so that the compound would probably exist in its ordered form at operating temperature. As is the case with many other ordered compounds, this composition has been found to disorder rapidly under irradiation, so that in a neutron field the compound probably exists in the form of the solid solution.

Note, however, that the Zircaloy-2 base with which the uranium is alloyed contains additions of Fe, Cr, and Ni, all of which are relatively insoluble in α-zirconium and exist in the form of intermetallic compounds with zirconium. It can be anticipated therefore that the uranium compound is associated with these intermetallics in some as yet unidentified form.

Prior to welding into subassemblies, the fuel plates are annealed at 1450°F. This temperature lies within the $\alpha + \beta$ range of the Zr-U binary system; consequently the structure of the fuel alloy after annealing consists of equiaxed α grains surrounded by a network of approximately 10 vol. % of the prior β phase which, upon cooling below 610°C, decom-

poses by a polynary eutectoid-peritectoid reaction to a two-phase mixture of α-zirconium and Zr-Fe-Cr-U compounds.

During subassembly fabrication the very edges of the fuel alloy (a thickness of 1/4 to 1/8 in. adjacent to the side plates) are heated to temperatures entirely in the β-phase of the fuel alloy. Therefore, the structure of the edge portions of the fuel alloy consists of a Widmanstätten type, acicular microstructure characteristic of β-quenched zirconium-base alloys. This structure is not changed appreciably by the subsequent annealing treatment at 550°C to which the seed subassemblies and clusters are subjected.

5–3.2 Properties of fuel alloy. Considerable work has been done on determining the mechanical properties of the fuel alloy and its corrosion resistance in high temperature water.

Zircaloy-2, the cladding material, demonstrates adequate corrosion resistance to high temperature water; however, when uranium is added to Zircaloy-2 the corrosion rate increases rapidly. Zircaloy-2 will exhibit a weight gain of 30 mg/dm^2 in 100 days exposure to 680°F water. The fuel alloy of the PWR Seed I composition shows a weight loss in 100 days exposure in 680°F water of about 1500 mg/dm^2. This alloy in the β-quenched condition shows an even higher corrosion rate, with a weight loss of about 5000 mg/dm^2 in 100 days exposure. Furthermore, the corrosion product on Zircaloy-2 is quite adherent and remains on the surface as a corrosion barrier, whereas the corrosion product of the fuel alloy ultimately sloughs from the surface, resulting in a linear loss in weight with corrosion time. Note, however, that in absolute terms the corrosion rate of the fuel alloy at 680°F is still quite small — equivalent to about 0.003 to 0.010 in. of metal attack per year.

As would be anticipated, addition of uranium to Zircaloy-2 markedly increases mechanical strength; both the 0.2% yield strength and ultimate strength are increased over those properties of Zircaloy-2 by more than 50%; e.g., at a temperature of 700°F the 0.2% yield strength of the fuel alloy is about 38,000 psi and the ultimate strength 47,000 psi. This improvement in mechanical properties is obtained at no sacrifice to elongation or reduction in area, the latter being in excess of 50% at 700°F. Unfortunately, no advantage can be taken of these properties in design since, as will be shown below, mechanical properties are adversely affected by irradiation. In fact, in the mechanical design of the fuel element the fuel alloy is not considered to contribute any strength to the structure.

5–3.3 Irradiation effects. As mentioned above, relatively minor amounts of uranium burnup in the fuel alloy are sufficient to reduce its ductility to negligible values; it has been found, in fact, that at less than 0.1 a/o

burnup no measurable plastic deformation can be noted in zirconium-uranium alloys when they are given post-irradiation tensile tests. Such behavior is characteristic of all metallic fuel materials; it has been noted not only in zirconium-base uranium alloys, but also in α- and γ-uranium-base alloys and aluminum-base alloys with uranium. It thus appears that nuclear fission sharply decreases the ductility of metallic materials. A tentative explanation for this behavior has been advanced, based upon the condensation of the noble gas atoms (xenon and krypton) on dislocations, thus impeding their mobility when stress is applied, and leading to early cracking as a consequence of such anchoring and piling up of dislocations.

It should, however, be noted that, in spite of the adverse affect of irradiation on mechanical properties, no separations between fuel and cladding occur at the bond, even upon plastic bending of the fuel elements after irradiation. Samples tested in this manner fail when cracks from the irradiated fuel are propagated into the cladding normal to the direction of the applied stress rather than along the fuel-clad interface. However, considerable bending is possible before the irradiated fuel element ruptures, due to the good resistance of Zircaloy-2 to crack propagation.

Of major concern with respect to using Zircaloy-base uranium alloys is the phenomenon of swelling. This appears to increase rapidly above a critical temperature, and can result in changing the volume of the fuel alloy as much as several hundred percent. For example, it has been found that in α-uranium this critical temperature for fuel swelling is approximately 500°C. Swelling is usually attributed to the precipitation and agglomeration of the insoluble noble gases formed in fission and to the resultant plastic deformation of the matrix from the imposed internal pressure stresses. At temperatures below the critical swelling temperature, these noble gases are either retained in the lattice or are prevented from expanding by the mechanical strength of the metallic matrix. Consequently, at these temperatures the resultant volume expansions noted upon fuel burnup are relatively small and are predictable; it is therefore desirable to hold maximum fuel temperatures below the critical swelling temperature of the fuel elements. Irradiations have been performed on Zr-U alloys with uranium contents of 3 to 41 w/o at temperatures from 100 to 1500°F attained both in-pile and by post-irradiation annealing. The increase in volume (expressed in percent change in volume per a/o alloy burnup) obtained at various irradiation or annealing temperatures is shown in Fig. 5-10. It may be noted that, at temperatures below about 900°F, the experimentally observed changes in volume are less than 6% per a/o burnup, whereas at higher temperatures large and variable amounts of fuel growth may be obtained.

It has been found experimentally, in the case of clad fuel plates, that this fuel volume change occurs primarily in the thickness direction of the plate;

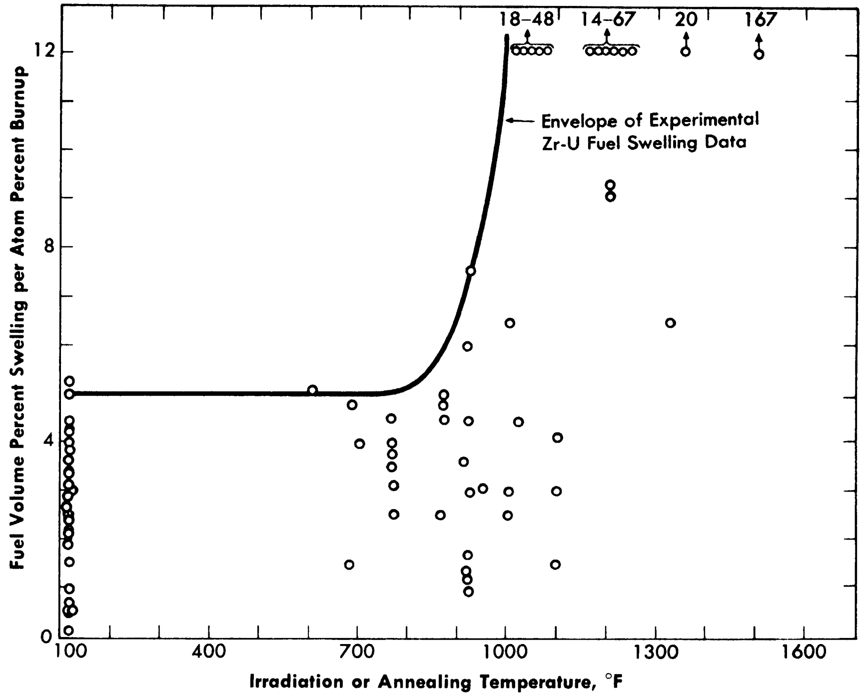


FIG. 5–10. Experimental observations of swelling of irradiated Zr-U alloy.

Thus for the anticipated maximum burnup in the Core I seed alloy (about 2a/o), a maximum increase in fuel thickness of 12% (or about 5 mils) would be expected for exposures below the critical swelling temperature. Since the maximum operating temperature of the Core I seed fuel alloy is expected to be 750°F or less, the change in thickness of the platelets is anticipated to be tolerable. Note that the plate configuration is peculiarly suited to conform to large volume changes of the fuel since large increases in volume can be obtained at relatively small cladding strains. Other shapes, such as cylinders, would have larger cladding strains for the same amount of volume change.

It has been shown that, in the case of the blanket fuel elements, volatile fission products can diffuse from the UO_2 matrix and, if cladding fails, contaminate the coolant; furthermore the entire contents of a defective blanket fuel element can contribute to the release of such fission products. However, in the case of the seed fuel alloy, large release of volatile fission products occurs only when exaggerated fuel swelling has taken place. It is probable that under these conditions the fuel alloy matrix cracks and the agglomerates or pockets of volatile fission products are able to communicate to the free surface. At irradiation temperatures below the critical swelling

temperature, the fission products are retained within the fuel matrix; with a cladding defect, fission products are released by recoils from the exposed fuel alloy surface and by corrosion of the exposed fuel alloy. Actual post-irradiation corrosion tests indicate that the metal loss for Zr-U fuel may be about ten times as high as the rate for unirradiated fuel material, which is about 0.003 to 0.010 in. per year.

The experiments outlined above, as well as operational experience with high integrity fuel elements of this type, are considered to have validated the selection of this material for the PWR seed.

5–3.4 Zirconium and Zircaloy-2. Zircaloy-2 was chosen as a cladding material for both the seed and blanket of PWR because of its low neutron absorption cross section, and adequate corrosion resistance and strength. Its thermal neutron absorption cross section is about 0.2 barn. By comparison, the absorption cross section of iron is about 2.5 barns.

The corrosion resistance of Zircaloy-2 stems from a protective ZrO_2 film that forms upon exposure to water or steam. The film is tenacious and does not spall off, as shown by corrosion tests at various temperatures for periods of over a year.

Zircaloy-2 has a density of 6.5 g/cm^3. The thermal conductivity of 8.2 Btu/hr-ft-°F is similar to that of austenitic stainless steels, and the thermal expansivity of 6.5×10^{-6}/°C is not too different from that of carbon steel. The minimum yield strength of Zircaloy-2 at 550°F is 17,000 psi. Total elongation at this temperature is about 50%.*

The primary sources of zirconium are the minerals zircon ($ZrSiO_4$) and baddeleyite (ZrO_2). These minerals are converted to chloride and reduced to zirconium metal with magnesium by the Kroll process. The resulting metal is called zirconium sponge because of its porous nature. In 1953, and again in 1956, the AEC established fixed price contracts for procuring zirconium sponge meeting a product specification.

Unalloyed zirconium can be used as a reactor material if it is sufficiently pure. Refinement of zirconium sponge by transfer and deposition on a hot wire using iodine as a carrier produced a material (called crystal bar because of its external hexagonal habit) which is pure enough for reactor use. However, such pure zirconium has low strength and loses its good corrosion resistance by pickup of small amounts of impurities such as nitrogen. Laboratory work was started to develop an alloy that could overcome these shortcomings. The result was the development of an alloy, named Zircaloy-2, containing 1.5% tin, 0.15% iron, 0.1% chromium, and 0.50% nickel. Tests indicate that this alloy is corrosion resistant even

*Information on Zircaloy-2 is contained in the book *Metallurgy of Zirconium* by Lustman and Kerze, published by McGraw-Hill in 1955.

with small amounts of impurities introduced during fabrication and that its yield strength is about three times that of crystal bar zirconium. In addition Zircaloy-2 permits using zirconium sponge directly, without the additional expense of the iodide deposition step.

5-3.5 Hafnium for control rod. The use of hafnium for the control rod in PWR is based on its high neutron absorption cross section and adequate corrosion resistance and strength. Hafnium is obtained as a byproduct in the process of making reactor grade zirconium, and is limited to about 2% of the amount of reactor grade zirconium produced. Hafnium is reduced by the Kroll process and refined by the iodide hot wire process. The resulting metal is workable and adequately strong and corrosion resistant. It is not necessary to add alloying agents to improve its properties. The minimum yield strength of hafnium at 500°F is 20,000 psi.

Basically, hafnium is very similar in chemical properties to zirconium. This is one reason that they occur together in ores; it also explains why they are difficult to separate. However, hafnium is almost twice as dense as zirconium, and has a melting point of 2000°C, as compared with 1850°C for zirconium.

5-4. Conclusions

Shippingport fuel element development culminated in the selection and validation of UO_2 as a satisfactory fuel material for the blanket. This conclusion was based on the experimental program undertaken to satisfy the criteria initially imposed:

(1) UO_2 exhibits a uranium loading density and neutron absorption cross section satisfactory for the nuclear design requirements of the reactor.

(2) No dimensional instability was exhibited by UO_2-containing fuel element samples at burnup levels to 25,000 Mwd/t.

(3) Fuel element samples have operated with defects in high temperature water systems at PWR reference heat fluxes to burnups greater than 10,000 Mwd/t for as long as 14 months with no evidence of corrosion or of gross release of radioactivity.

(4) No evidence of progressive failure has been noted in experimental fuel element assemblies deliberately fabricated to produce such a condition in in-pile irradiation tests.

The properties listed above indicate that metal-clad, ceramic-fueled fuel elements may be applicable with other cladding, fuel materials, and coolants. In fact, this concept is now being investigated by other organizations for the fuel element design for reactors employing: (1) aluminum alloy and stainless steel cladding in addition to zirconium, (2) ThO_2 fuel as well as UO_2, and (3) boiling water and sodium coolants as well as pressurized water.

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CHAPTER 6

CORE MANUFACTURING

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This chapter describes both the manufacturing processes employed to fabricate the core components and the quality evaluation methods used on the finished products. Careful engineering development preceded, and in many instances went along concurrently with, manufacture of the core components.

The core design was necessarily conservative because previous operating experience was lacking, and that lack imposed stringent dimensional control and equally stringent requirements on joining method integrity and reliability. Thus, unusual problems had to be faced in fabricating all core components. They ranged from establishing a mass production method to convert uranium dioxide powder into dense oxide fuel, to welding Zircaloy-2, hafnium, and stainless steel components. To solve them, new processing methods and assembly techniques had to be evolved.

6–1. THE BLANKET

6–1.1 Terminology. The blanket region of the core contains 113 fuel units, termed *blanket assemblies.* Forty-five blanket assemblies are inside the annular seed, 68 outside the seed. Each blanket assembly contains seven subunits, termed *fuel rod bundles,* confined within a Zircaloy-2 housing, referred to as a *shell.* The shell serves as a structural member as well as a water flow director unit. Each shell has a bottom butt-welded transition piece of Zircaloy-2; this combination is referred to as the *shell component.* Stainless steel hardware is used for both the latching mechanism at the top end and the spring retaining mechanism at the bottom (transition) end. The combination of a welded full length water sampling tube confined within the *shell component* is referred to as the *shell assembly.* The combination of the shell assembly, stainless steel hardware, a water sampling rake, and Inconel springs, together with the fuel bundles, forms a complete *blanket assembly* as described in Chapter 4.

Each of the seven fuel rod bundles found in a blanket assembly contains 120 *fuel rods.* The fuel rod is the basic heat generating unit of the blanket. Contained within each fuel rod is the natural uranium dioxide (UO_2) fuel in the form of 26 right circular cylinders, referred to as *pellets.*

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6–1.2 The fuel component (the UO_2 pellet). There were four steps in the fuel component manufacturing process: agglomeration, compaction, sintering, and grinding.

Agglomeration converted "as-received" nonflowing UO_2 powder into free flowing granules. ("As-received" UO_2 refers to the powdered raw material produced by the Mallinckrodt Chemical Works by the hydrogen reduction of UO_3 obtained from mineral sources.) The brown colored UO_2 powder had an apparent density of 3.0 g/cm^3, contained greater than 50 w/o fines (through-325-mesh particles), and consequently was not free flowing. In the agglomeration step the very small particles were bound together by a water soluble organic binder, polyvinyl alcohol (PVA). Sterotex, a hydrogenerated vegetable oil product, was then blended into the batch to provide lubricity and increase compactibility. After blending, the granular UO_2 was ready for charging to the compacting press. Since the basis of manufacture was cold pressing, producing free flowing granules was the key operation in obtaining uniformity of volumetric die fill and hence uniformly pressed compacts.

Compaction is the consolidation of the loose granules into a right circular cylindrical shape by cold pressing. The pressure used was 125 tons/in^2 and resulted in a green (or "as-pressed") density of 73 to 74% of theoretical. The pressed compact, 0.4 in. in diameter by 0.4 in. long, had good green strength and could withstand reasonable handling. The compacting was performed on Stokes 280-G presses of 100 ton capacity at the rate of 20 compacts per minute. These double acting automatic mechanical presses were equipped with hydraulic pressure equalizers to insure uniform compacting pressure. Double action pressure was exerted on the granular material from above and below by a moving die table, which descended during compression at half the speed of the upper punch. Both the die liner and the upper and lower rams or punches were constructed from solid tungsten carbide, an important feature enabling the heavy compaction pressures to be used on a production basis.

Sintering was the heat treatment used to achieve high density of the cold pressed compacts. As a preliminary step before sintering, the compacts were preheated in a separate furnace to remove the volatile hydrocarbons introduced by the organic binder (PVA) and the lubricant (Sterotex). This operation did not influence the final sintered density, but avoided contaminating the sintering furnace.

Because of the many UO_2 compacts required for Core I, large capacity molybdenum wound hydrogen sintering furnaces were used. The production sintering process converted the 73 to 74% theoretical density "as pressed" compacts to 93 to 95% sintered pellets by applying a temperature of 1675°C for a period of eight hours.

Grinding was the final step used to obtain precise dimensions. The two step operation was done with conventional grinders. The diameter of the

pellet was ground first on a centerless grinder; then the required length was ground on a double disc end grinder. Both grinders were fitted with silicon carbide abrasive wheels. Water containing a rust inhibitor was used as the coolant in both operations. Pellets were ground at the rate of 25 per minute. The maximum material removed per pass was 0.015 in. The diameter was controlled to ± 0.0005 in. with a maximum ovality of 0.0001 in. Squareness of the ends was held within 0.002 in.; stacked pellet length (9.084 in. for 26 pellets) was held to ± 0.023 in.

The UO_2 fuel component was assessed for quality on the basis of dimensional, density, and corrosion resistance specifications. Density was checked by gravimetric and dimensional measurements and verified by water displacement measurements. The sintered and ground pellets were checked for corrosion resistance in 750°F steam at 2000 psig for 20 hours. Not a single corrosion failure, as evidenced by physical change, was found among the 166,000 pellets tested during the entire manufacturing period. These 166,000 pellets amounted to a 4% random sample of the total fuel component production.

6–1.3 The fuel rod element. The fuel rod, an unbonded type of element, consists of UO_2 fuel contained within a length of Zircaloy-2 seamless tubing. Each rod was fabricated from 26 UO_2 pellets, a cladding component, and an end cap as illustrated in Fig. 6–1. A length of Zircaloy-2 seamless tubing and one end cap were fusion welded together to form the cladding component.

The bulk of the Zircaloy-2 seamless tubing was made by one commercial tubing producer, using essentially a two-part process. The first part consisted of the hot extrusion of a hollow billet to a tube shell. The hollow billets were prepared by hot extruding 12 in. diameter ingots of Zircaloy-2 to $7\frac{1}{4}$-in.-diameter bars. The bar was machined on the outside diameter and a hole was bored which yielded a hollow billet having nominal dimensions of 6.75 in. OD by 1.82 in. ID. These billets were jacketed in steel and copper and extruded at a temperature not exceeding 1550°F. These two aspects of processing are unique to Zircaloy material. The jacketing prevents oxidation of the Zircaloy and achieves better material flow in extrusion. Maintaining the proper temperature of extrusion is important to minimize or prevent reaction between the Zircaloy and the jacketing material. The nominal dimensions of the extruded tube shell were $2\frac{1}{2}$ in. OD by 1/4 in. wall thickness. Dejacketing was done in nitric acid; the outside and inside surfaces of the stripped and pickled tube shell were mechanically conditioned to eliminate any mechanical defects and/or alloy contamination.

The second part of the process consisted of cold working the hot extruded tube shell to finished size. Tube reducing and mandrel and plug drawing gave a maximum reduction of 45%. Intermediate nitric-hydrofluoric acid

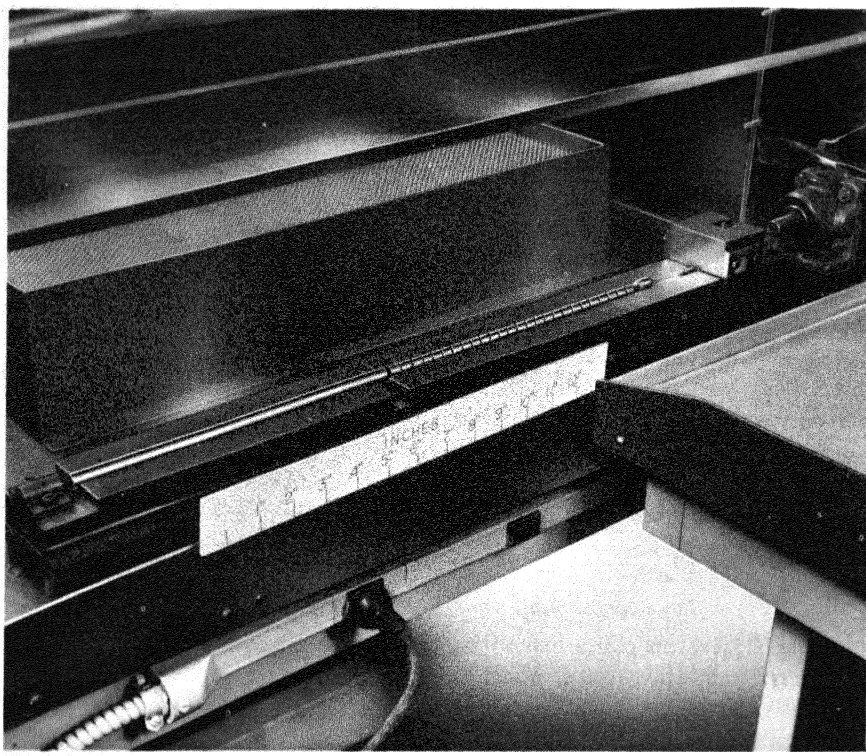

FIG. 6–1. Loading of UO_2 into fuel rod, showing (in vee block fixture from left to right) cladding component, 26 pellets, and second end cap. Although the second end cap is shown here, it is not inserted until the cladding component has been evacuated and then filled with high purity helium.

pickling and vacuum annealing at 1600°F for one hour were employed in conjunction with this cold working. Nitrogen pickup was prevented during annealing by the combination of high vacuum and absolutely clean tubing surfaces. (Nitrogen is detrimental to the corrosion resistance of Zircaloy-2.) The tube was then straightened, centerless-belt polished on the outside, and sandblasted on the inside. The finished mill length of seamless tubing was subjected to a 5000 psi hydrostatic pressure proof test and checked for flaws with an eddy current detection device. The nominal dimensions of the finished tubing were 0.420 in. OD by 0.028 in. wall thickness.

After the first end closure weld was made, its integrity was checked by applying a radiographic (porosity detection) test and a 300-psi helium leak test.

The cladding component was loaded with fuel pellets in a vee block fixture. It was then evacuated and filled with high purity helium. The

second end cap was then inserted and welded in place in a hermetically sealed chamber. The second weld was tested in the same manner as the first. The welding parameters employed for the first end closure were 38 to 40 amp DC, 15 to 17 volts, at 15 rpm for 4.5 sec (1.1 revolution), with a thoriated tungsten electrode. The variation for the second end closure consisted of making a two revolution weld and employing a taper control on the welding current. A corrosion test in 750°F steam at 1500 psig for three days was made on the welded fuel rod after cleaning and pickling, as an added control measure. The finished nominal dimensions of a fuel rod in a bundle are $10\frac{1}{4}$ in. long by 0.413 in. OD. The nominal fuel dimensions are 9.084 in. long (for a stack of 26 pellets) by 0.3575 in. OD. The minimum tubular cladding thickness in the finished fuel rod bundle is 0.0225 in.

6-1.4 The fuel rod bundle. The fuel rod bundle (Fig. 4-15) consists of 120 fuel rods welded into two drilled Zircaloy-2 plates (referred to as tube sheets) which space and support the rods in a square lattice.

There were four steps in the manufacture of fuel rod bundles: (1) assembly, (2) fusion welding, (3) annealing, and (4) machining.

In the assembly operation, the rods were inserted into one tube sheet along with spacers of Ti-Namel (a titanium killed steel) at three locations uniformly spaced along the longitudinal axis of the bundle. The Ti-Namel spacers were placed 3/8 in. from each tube sheet and also at the center, to aid in achieving specified rod spacing in the finished bundle. With all the rods in one tube sheet and all the spacers in place, the second tube sheet was pressed on lightly. After checking for squareness, the rods were lightly tack welded in air at both tube sheets to restrain any movement in handling.

The fusion welding was done in a hermetically sealed chamber of the type shown in Fig. 6-2. The chamber was initially evacuated to less than 0.03 micron pressure and then filled with one atmosphere of high purity helium. A weld sequence based on theoretical stress analysis and experimental verification was used to make the 120 welds on each tube sheet. This weld sequence, as illustrated in Fig. 6-3, was all important in controlling fuel rod spacings within close limits. A thoriated tungsten electrode rotating at 30 rpm was used. The current used was 180 amperes. The time of current application (i.e., the heat input) varied from 12.5 sec to 14 sec, depending on the fuel rod location.

The fusion welded bundle was vacuum annealed at 1275°F for three hours with the Ti-Namel spacers in place, to remove locked-in welding stresses. In-service movement of the rods in the bundle was thus virtually eliminated. Four sets of stainless steel and molybdenum restraining clamps were strategically placed along the longitudinal axis of the bundle to control dimensions during the stress relief anneal. The high temperature anneal

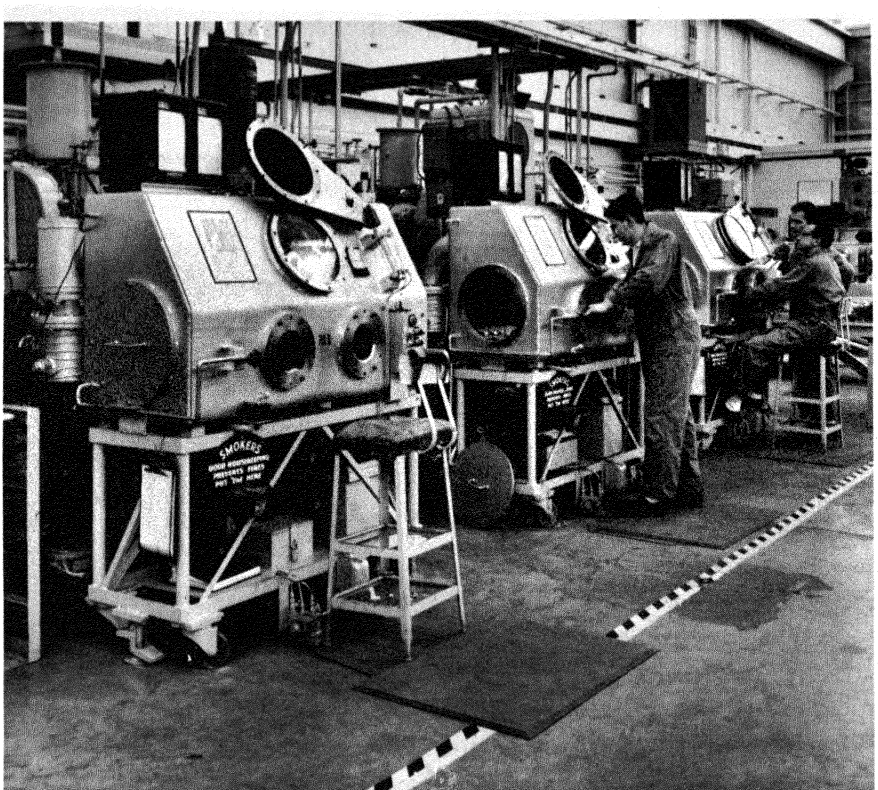

FIG. 6–2. Hermetically sealed welding chambers.

also allowed any surface contamination on the Zircaloy to be diffused; hence the final corrosion resistance of the bundle was not impaired. After annealing, the Ti-Namel spacers were dissolved with nitric acid.

The final operation consisted of machining the bundle to specified dimensions, using special fixtures on standard machine tools. The tube sheet faces were milled to clean up the surfaces and achieve the specified length. Next, water flow holes were drilled in both tube sheets. Finally, the tube sheet edges were milled to obtain proper width and proper distance between the outer rods and the tube sheet edges.

The fuel rod bundle was checked for compliance with specifications on dimensions, corrosion resistance, and fuel rod cladding integrity. Fuel rod spacing, distance from outer rod to tube sheet edge, and over-all length were measured. The machined bundle was cleaned and then corrosion tested in 680°F water at 2705 psig for three days. A 300-psi helium leak test was applied to the corrosion tested fuel rod bundle as a final check for leaking elements.

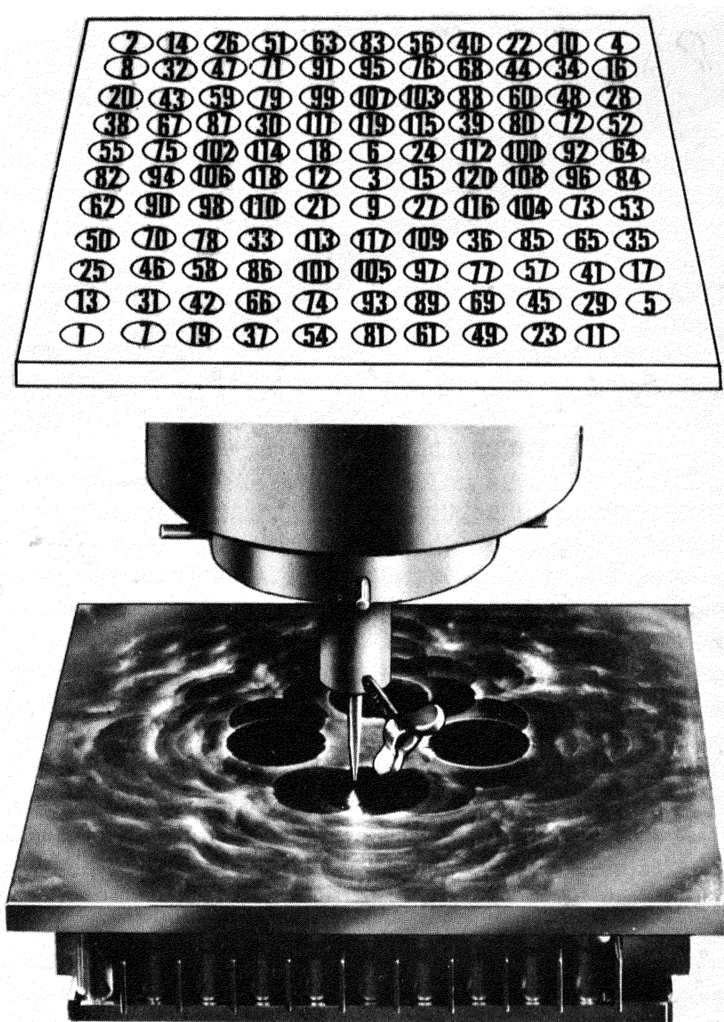

FIG. 6-3. Fuel rod bundle welding sequence.

6-1.5 The shell assembly. The shell assembly (Fig. 6-4) consists of the shell, a transition piece, and the sampling tube. This is an all Zircaloy-2 unit which was fabricated by fusion welding the transition piece and sampling tube to the shell. The shell, main component of the shell assembly, is a tubular structure about $5\frac{1}{2}$ in. square and $105\frac{1}{4}$ in. long. The shell was produced in three steps: (1) tack welding, (2) fusion seam welding, and (3) machining.

Four strips of Zircaloy-2, each approximately 1/4 in. thick by $5\frac{1}{2}$ in. wide by 9 ft long, were machined from hot rolled and annealed stock. The

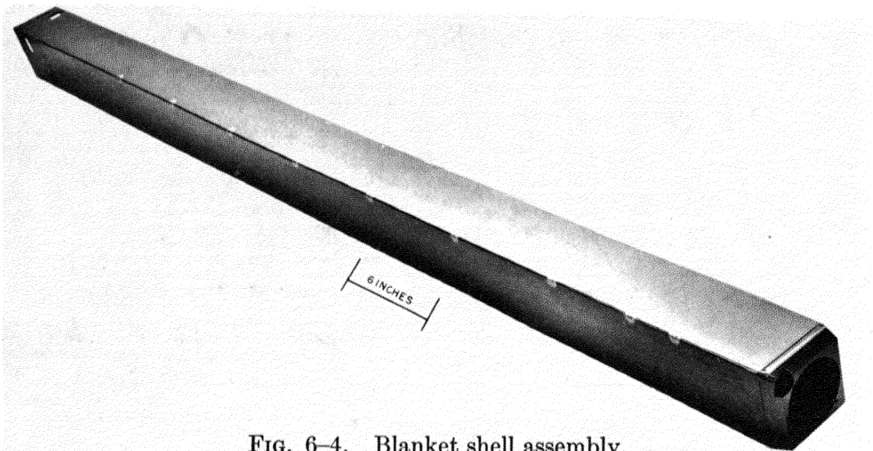

FIG. 6–4. Blanket shell assembly.

strips were positioned in a fixture and tack welded in air at approximately 3-in. intervals along each of the four corners. Two internal fixtures were then inserted into the shell to ready it for fusion welding in helium in a hermetically sealed chamber. The internal welding fixture consisted essentially of copper back-up bars for the seam weld. These were properly supported to provide desired dimensional control of the shell. The as-welded outside dimensions were held within a critical reference envelope. The as-welded inside dimensions were held to close tolerances, since there is a 0.005-in. to 0.025-in. range of clearance between the bundle tube sheet and the inside of the shell. The fusion welding parameters employed were 215 to 225 amp DC, 15 to 17 volts, at 9.5 to 10.5 in/min, using a thoriated tungsten electrode. The machining operations on the shell were limited to external seam weld cleanup on a milling machine and internal weld grinding where necessary.

After the shell was seam-welded, machined externally, and ground internally when required, it was rough-cut to length. One end of the shell was face-milled before the transition piece was fusion welded to form the shell component. The transition piece is a square plate approximately 7/8 in. thick forming the bottom of the shell assembly. Two parallel surfaces on the transition piece were machined after welding. The top of the shell was then machined parallel to these surfaces. The latch mechanism holes on the shell sides and the transition piece bore were machined to final size after corrosion testing of the shell component in 680°F water at 2705 psig for three days. The sampling tube, consisting of heavy wall Zircaloy-2 tubing, was similarly corrosion tested and then welded into one corner of the shell component to complete the shell assembly. The shell assembly was evaluated for quality by checking corrosion resistance, dimensional requirements, weld penetration, and surface appearance.

6-2. The Seed

6-2.1 Terminology. The seed region of the core contains 32 fuel units, termed *clusters*. The nominal over-all dimensions of a cluster are $5\frac{1}{2}$ in. square by $97\frac{1}{4}$ in. long. Each cluster consists of four subassemblies joined together by welding to four Zircaloy-2 spacers to form a cruciform-shaped channel in the center of the cluster. Each standard subassembly consists of 15 fuel elements and two non-fuel-bearing end plates. A *source cluster* contains one spacer which has been machined to receive a neutron source. A completed cluster has an *extension bracket* welded to the top and a *transition section* welded to the bottom. Each cluster also contains an orifice assembly to restrict the water flow through the cruciform-shaped channel. The control rod operates in this channel.

6-2.2 The fuel alloy. The fuel alloy was manufactured by means of an inert atmosphere arc melting process. A charge — consisting of enriched uranium pellets and a blend of Zircaloy-2 and crystal bar zirconium, plus necessary alloying pellets — was melted into a 4-in. diameter, 20-lb ingot in a furnace back filled to a slight positive pressure with five parts helium and one part argon. After melting, the ingot was encased in a copper jacket, heated to 1650°F in an electric furnace, and forged into a billet approximately $6\frac{1}{2}$ in. wide and 3/4 in. thick. The billet was then dejacketed and rejacketed for rolling. This assembly was heated to 1550°F and hot rolled into strip approximately 0.135 in. thick. After dejacketing, the strip was cleaned and chopped into 1/8 in. by 1/8 in. pellets. This chopped alloy material was used as the charge for a second melting.

The second operation was identical to the first except that the resulting ingot was $2\frac{1}{2}$ in. in diameter. This $2\frac{1}{2}$-in. diameter ingot formed the electrode for the third and final consumable melting to form a 4-in. diameter ingot, which was forged and rolled as previously described. The three melting operations gave assurance of the complete homogeneity of the fuel alloy. After hot rolling, the strip was cleaned by pickling in a mixture of nitric and hydrofluoric acids, cold rolled to approximately 0.123 in. thick, and pickled to final thickness. The strip was then sheared into twelve filler blanks which were machined to final dimensions for fuel element assembly.

To assure a quality product, inspections were performed at critical stages throughout the process. Before fillers were released for fuel element assembly, they were visually inspected for external defects and radiographed for internal defects. The dimensions, weight, and uranium loading were carefully checked. Uranium homogeneity checks were made on the strip as well as on the fillers machined from the strip.

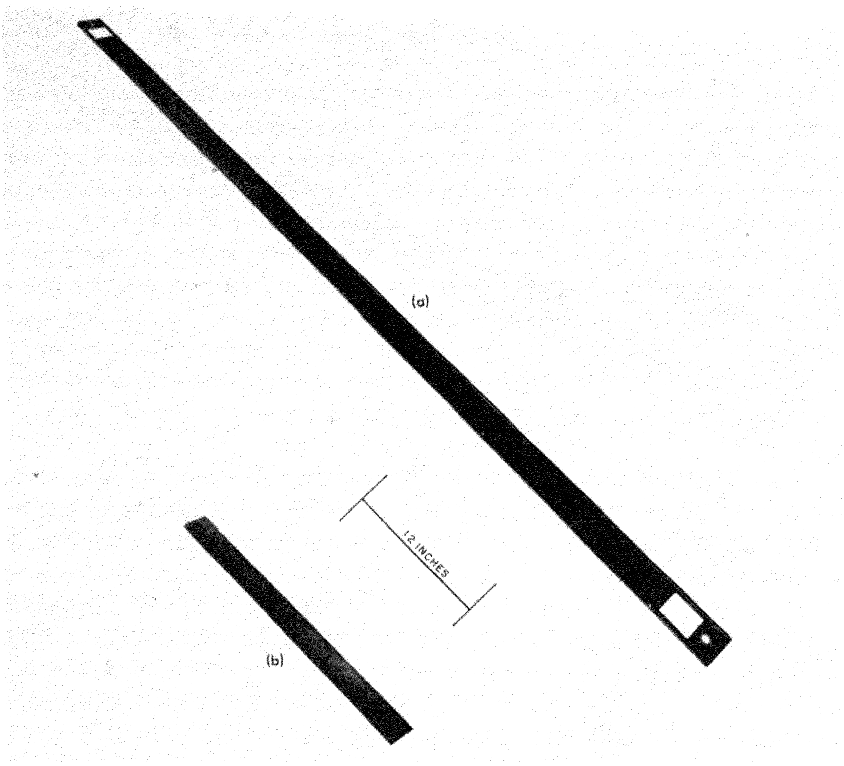

FIG. 6–5. (a) Seed fuel plate after rolling and blanking. (b) Cover plate component before rolling.

6–2.3 The fuel plate element. The fuel plate elements utilize recessed Zircaloy-2 cover plates as envelopes for the fuel filler. Since the rolling process for fuel elements consisted of a 3:1 reduction schedule, the plate components were made to one-third the final fuel element length and three times the final thickness (see Fig. 6–5). To obtain the required fuel element I-beam cross section, design flange strips were positioned on both sides of the cover plates by means of steel spacers during the fuel element assembly operation. Because Zircaloy-2 oxidizes readily, components of this material were enclosed in a protective jacket made from Ti-Namel during the hot rolling operation. The Zircaloy-2 cover plate and fuel alloy filler components were cleaned and assembled into the Ti-Namel steel jacket, which was then welded together and evacuated. After each assembly was found to be vacuum tight by means of a helium probe, the tube of the assembly was sealed by a forge welding operation. The assembly was then hot rolled.

The 3:1 rolling schedule consisted of seven hot rolling passes and a final cold rolling pass to achieve the specified fuel element length. Rolling was

preceded by a heating period to bring the assemblies up to the rolling temperature of 1450°F. Between passes, the assemblies were reheated to this temperature, the assemblies being alternated end for end and top to bottom during rolling to minimize directional effects of rolling. An anneal after the last pass concluded the hot working schedule. Following this anneal, the fuel element was removed from the Ti-Namel jacket and processed for assembly into a subassembly.

The processing of the fuel elements subsequent to hot rolling and Ti-Namel dejacketing started with roller leveling and straightening. After this step, the fuel elements were nondestructively tested for any unbonded areas which would impair the integrity of the finished element. After each fuel plate element passed the bond test, the entire element was x-rayed for core and cladding integrity and fuel filler location. This location was important in determining reference positions for subsequent machining and blanking operations in which the fuel element was brought to proper width and contour. In the blanking operation a test coupon of non-fuel-bearing Zircaloy-2 was obtained from each end of the element (see Fig. 6-5). At least one of these coupons was corrosion tested for three days and for 14 days in 750°F steam. Both the coupon and the fuel element had to show a black lustrous film free of white oxide deposits to be acceptable. The test coupon was also used in metallographic studies to examine the quality of bonding between the Zircaloy-2 cover plates and the fuel alloy filler.

After the fuel element was corrosion tested and inspected, the black oxide film was removed and the element given a final cleaning. The latter was essentially a nitric-hydrofluoric acid pickling, water rinsing, and drying operation. The final fuel plate element was then ready for subassembly welding.

6-2.4 The subassembly. Each standard subassembly consists of 15 fuel elements and two non-fuel-bearing end plates. These components were joined by fusion welding along the flange seams of the fuel elements and end plates to form a unit approximately $2\frac{1}{2}$ in. square in cross section. Figure 6-6 shows a subassembly unit after welding and before machining has been completed. The non-fuel-bearing (Zircaloy-2) remnants of the individual plate fuel elements from which corrosion test coupons were previously blanked (see Fig. 6-5) may be seen in the bottom end of the subassembly illustrated in Fig. 6-6. Steel spacers were used in the 14 channels between the fuel elements and also in the two end channels between fuel elements and end plates. These spacers were employed to restrict weld shrinkage and keep the channel spacing within design limitations. The spacers were removed chemically before proceeding with any further processing of the subassembly for cluster construction. Following

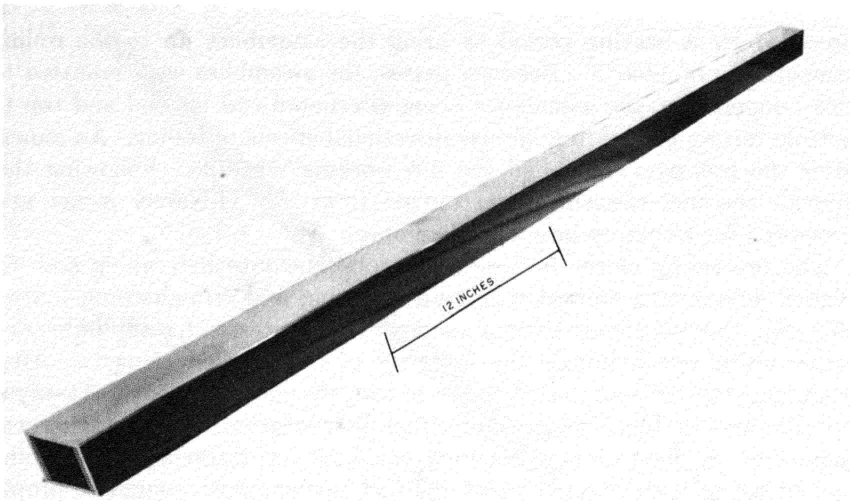

FIG. 6-6. Seed subassembly.

an evaluation and acceptance of the subassemblies for weld penetration and channel spacing requirements, the subassemblies were machined for assembly into a cluster.

All welding of Zircaloy-2 fuel elements and end plates was conducted in an inert atmosphere chamber (similar to the chambers shown in Fig. 6-2), utilizing helium. To obtain the purest helium atmosphere possible, the welding chamber was evacuated by means of a system of mechanical and diffusion pumps to a pressure of 0.03 micron before admitting the 99.997% pure helium gas. A slight positive pressure of helium was maintained in the chamber during welding. Because helium is a strategic resource, studies are underway to use argon for this purpose in all future work in lieu of helium. The substitution is not straightforward, however, and it is not yet clear under what conditions, if any, argon can be used.

A direct current source of energy with automatic voltage control was employed to maintain constant heat input to the components being welded. Nonconsumable, thoriated tungsten electrodes were held in an electrode holder, which was fixed longitudinally in the center portion of the welding chamber. Transverse or longitudinal movement of the work piece was maintained at a specified speed by a Graham drive mechanism. The fusion welding parameters used were 110 to 120 amp DC and 16 to 17 volts at a speed of 10 in./min.

6-2.5 The cluster assembly. The fuel cluster unit is composed of four subassemblies welded together with spacers to form the cruciform-shaped control rod passage. The spacers were machined from solid bars of Zircaloy. During assembly and welding the spacing of the control rod channel at

the center of the cluster was controlled by a pure aluminum* cruciform spacer bar. Once the cluster was assembled and tack welded, a tensile stress was applied to the ends of the annealed aluminum cross, thereby reducing its cross section to allow for subsequent weld shrinkage. The exceptional ductility and uniform reduction of area of pure aluminum made this material ideally suited for this application. After assembly welding, the cluster was annealed to stress relieve the welds and further anneal the aluminum spacer. The spacer bar was then removed by stressing to fracture. Zircaloy foil was used to protect the surfaces of the control rod channels from being scratched or otherwise marred during the reduction and later removal of the aluminum spacer.

The upper extension bracket was fabricated by welding; it is a 17$\frac{3}{8}$ in. long by 5$\frac{1}{2}$ in. square shell having 0.170 in. thick walls. The shell has four 1 by 1$\frac{1}{2}$ by 6 in. bearing ledges on the inside and eight rubbing shoes, 1/8 by 5/8 by 1$\frac{7}{8}$ in., on the outside. The walls of the bracket were fabricated by welding the heavy rubbing shoe and bearing ledge sections into a thin wall plate. This conserved Zircaloy which would have been lost in machining from a heavy plate—a plate approximately 2 in. thick would have been required for each side. Mandrels were used during welding and annealing to maintain control of dimensions and prevent distortion. The transition sections for the bottom of the clusters were initially formed by die forging and were then contour machined.

In welding the upper extension bracket and transition section to the fuel cluster unit, weld backups controlled penetration in the joints between these components. Again, special welding conditions and sequences were employed to assure weld penetration and minimize distortion.

In the cluster fusion welding, the current used was varied from 150 to 185 amp DC, depending on the type of jointure (assembly of cluster, joining of extension bracket to cluster, joining of transition piece to cluster). Voltage and speed remained fixed at 16 to 17 volts and 8$\frac{1}{2}$ in./min.

Subsequent to the extension bracket and transition piece welding, the cluster was machined, inspected, corrosion tested for three days in 680°F water, and final-inspected. A completed, corrosion tested cluster ready for core assembly is shown in Fig. 6–7.

The orifice unit is held in place by retainer pins press fitted into it and fusion welded into the spacer.

Before a cluster was released for assembly into the core, the transition section was selectively fitted with a stainless steel spring bearing ledge to assure proper clearance and operation.

Source clusters were fabricated in the same manner as standard clusters except for the one spacer to house the neutron source. This spacer contains

* Commercially pure aluminum, designation 2S-O.

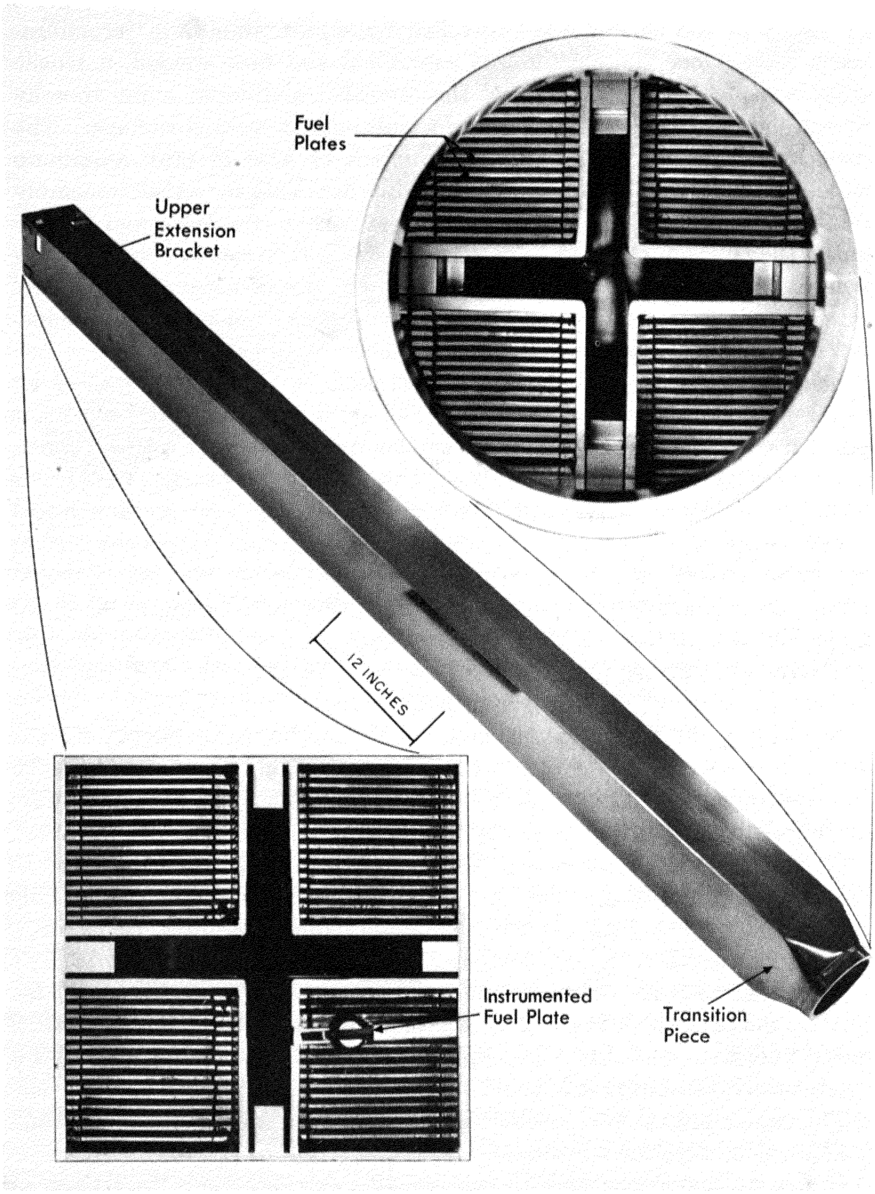

FIG. 6–7. Composite view of seed cluster.

a $\frac{1}{4}$-in.-diameter hole approximately 20 in. long inclined at an angle of one degree to the vertical axis of the spacer. This hole permits cooling water to flow around the source. The recess to house the source holder was machined into this spacer during intermediate machining.

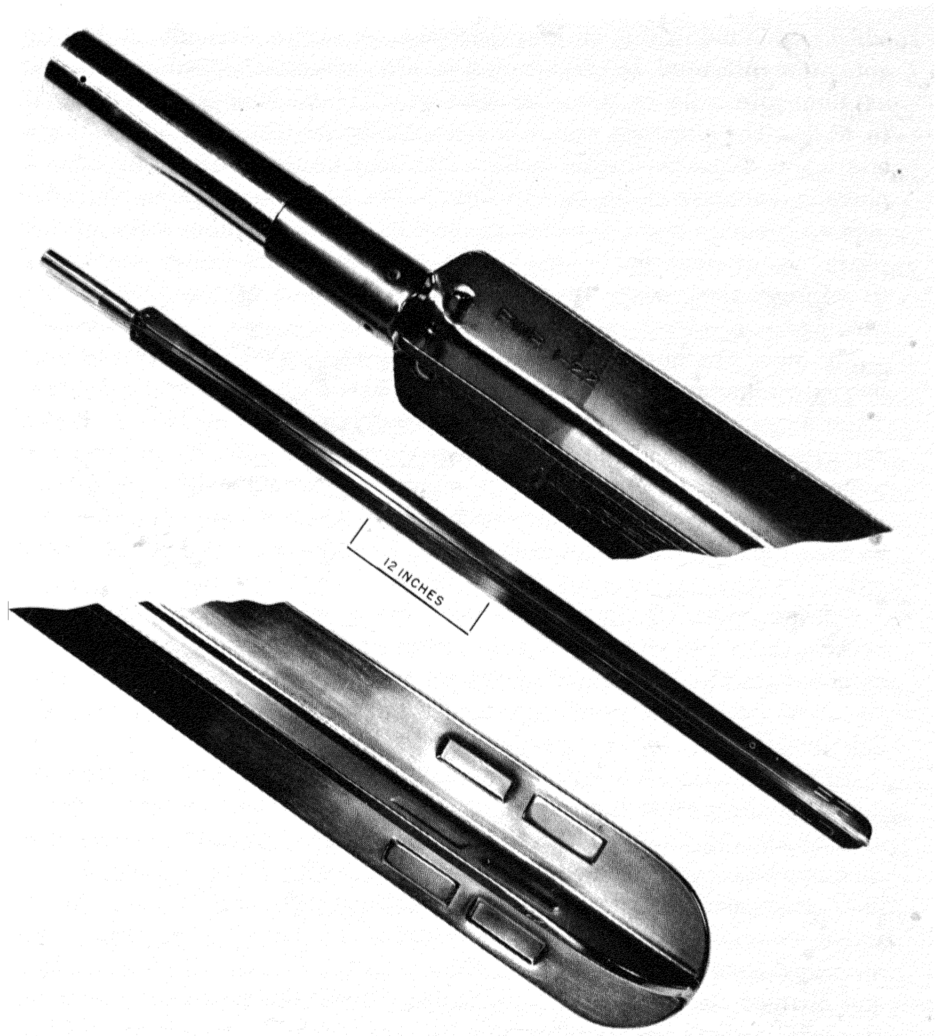

FIG. 6-8. Hafnium control rod.

6-3. The Control Rods

The control rods, which are inserted into the 32 cruciform passages in the seed clusters, consist of hafnium cruciforms with Zircaloy-2 adapters. The over-all length of the rods is $83\frac{3}{4}$ in., the hafnium portion being $71\frac{1}{2}$ in. long (see Fig. 6-8). The blade thickness of both the hafnium and Zircaloy-2 portions is 0.226 in. and the cruciform span is $3\frac{3}{8}$ in.

The hafnium used in these rods was in the form of 100% iodide crystal bar which was corrosion tested for 14 days in 750°F steam. Any bars

which exhibited white or gray corrosion products were rejected. Ingots were prepared by first tungsten arc melting chopped crystal bar hafnium into a $2\frac{1}{2}$ in. diameter electrode, $21\frac{1}{2}$ in. long and weighing 46 to 48 lb. The electrode was then consumably arc melted to a 4-in.-diameter ingot. Nonconsumable melting was done at a pressure of one atmosphere in a mixture of five parts helium to one part argon. The consumable melting pressure was ten inches of mercury in a 1:1 helium-argon atmosphere. After consumable arc melting, the ingots were fusion conditioned to improve the surface by shielded inert gas nonconsumable electrode welding.

The ingot was fabricated into strip by heating at 1900°F in an air furnace flooded with argon and forging into a billet about 1 in. thick and 4 in. wide. These forged billets were then hot rolled at 1700°F, 10 to 15% reduction per pass being made until they were approximately 0.285 in. thick. After rolling, the strip was sandblasted to remove oxide scale and then machined all over to obtain strips $71\frac{3}{4}$ in. by $3\frac{5}{8}$ in. by 0.236 in. At this time a radius was machined on one end of each strip to form the lower end of the control rod.

Spacing shoes, shown in the lower view of Fig. 6–8, were formed by heating the radius ends of the hafnium strips in sodium-potassium-lithium carbonate salt at 1650°F and offsetting the blade with a forge press. After the shoes were thus formed, the strips were cleaned by sandblasting and a light pickling in a nitric-hydrofluoric acid aqueous solution.

To machine the strip for cruciform assembly, half of the strips were slit down their long centerline with a 0.166-in. milling cutter to form two pieces each about $1\frac{47}{64}$ in. in width. Slots 0.030 in. deep by 0.224 in. wide were then machined on each side of the remaining strips along the long centerline. Degreasing and pickling of these components preceded assembly.

The cruciform was formed by inserting the two $1\frac{47}{64}$ in. wide pieces machined from the first strip into the slots machined into the second strip and aligning the pieces with special jigs. While the assembled cruciform was held in these jigs, the three pieces were tack welded together, using a shielded inert gas nonconsumable electrode in air. The cruciform was then placed in another jig and inserted into a welding box.

After evacuating the welding box and back filling with helium, root welds were made in each quadrant of the cruciform at 150 amp, 16 volts, and a speed of 6 in./min. To insure adequate penetration at the cruciform root, a second welding pass was made along each quadrant with the current increased to 170 amp.

To stress relieve the cruciform after welding, it was vacuum annealed for one hour at 870°C. A complete dimensional survey was made after annealing and the cruciform was straightened to remove any twisting, camber, or bowing. The top end of the cruciform, shown in the upper view

of Fig. 6–8, was machined perpendicular to its centerline; a 3/16-in. hole was center drilled in the top to promote a complete welding of the hafnium cruciform to the Zircaloy adapter and insure that there would be no unwelded hafnium-Zircaloy interface which could give rise to stresses.

The adapter was machined from a solid piece of Zircaloy-2, no special techniques being involved. To attach the adapter to the hafnium cruciform, both pieces were aligned in a special fixture and tack welded in air. The assembly was placed in a welding box, where a thoriated tungsten electrode was held at the center of each of the four root (hafnium to Zircaloy-2) junctions for a period of 5 to 10 sec. Radial blades were welded at 160 amp, 16 volts, and a speed of 6.1 in./min. All welds were started at the root of the cruciform and proceeded to the edge of the blade. One welding pass was made on each side of the blade—a total of eight passes in all. Run-out tabs were used at the edge of each blade to prevent burning.

After welding and removal of the run-out tabs, the control rod was vacuum annealed for one hour at 700°C to remove all welding stresses. The rod was then surveyed and straightened and the centerline for boring the adapter was determined. This was an extremely important operation, since the mean free path of the rod was determined by the accuracy and location of this centerline.

To ensure against distortion of the rod due to clamping during machining, a machining fixture was designed in which the rod was cast in "Cerralow" (composition 49.4% bismuth, 18% lead, 11.6% tin, and 21% indium). This low melting (58°C) alloy held the rod firmly in place, yet caused no strains to be set up in the rod during machining. In this one fixture, the adapter was bored and tapped, the cruciform width or span machined, and the spacing shoes machined to the height required by the mean free path and established centerline. Only the blades which lay horizontal were machined; consequently the rod had to be recast in the fixture four separate times so that all spacing shoes could be machined. After machining, the rod was degreased, pickled, then corrosion tested for one day in 750°F steam. The final operation consisted of threading a stainless steel adapter into the Zircaloy adapter, pinning it with a Zircaloy taper pin, and welding over the taper pin.

6–4. The Structural Supports

6–4.1 Terminology. Structural support is an over-all term; it refers not only to those parts of the core which support the fuel components throughout service life, but also to those non-fuel components which fulfill functional requirements other than support. For example, the control rod linkages between the drive mechanisms outside the reactor vessel and the control rods proper are classed as structural supports, but their function is to guide and align the actuated control rods.

The principal assembly classed as a structural support is the *core cage.* This consists of three components: the *core barrel,* the *top grid,* and the main supporting member, the *bottom support plate.* The top grid and bottom support plate support and align the core fuel units for proper passage of the control rods and for correct spacing of the fuel units. The top grid and bottom support plate are pinned to the core barrel. The static load is transmitted to the core barrel flange and thence to the reactor vessel ledge. The operating hydraulic load is an upward thrust which is transmitted to the *hold-down barrel* and to the reactor vessel head. Other major assemblies in the core structural support category are the *control rod shrouds and linkages* and the *instrumentation systems.* As discussed in Chapter 4, the seed-and-blanket fuel unit assemblies contain latching devices, flow metering units, and other stainless steel parts that are classed as structural supports.

6–4.2 The top grid and bottom support plate. The top grid and bottom support plate are both of grid type construction, containing square openings with light walled separating ribs and measuring approximately $7\frac{1}{2}$ ft in diameter by 1 ft in thickness. They support and align the fuel units. A heavy ring encompasses each grid. Peripheral holes in the top grid are irregular in shape and act as water passages, while those in the bottom support plate are blind holes. The main distinction between the two plates is that the bottom support plate has a complete top cover with holes centered on grid boxes to accommodate adapters which fit into the fuel units.

Both the top grid and the bottom support plate were fabricated as weldments from extruded cruciform shaped bars. This method was chosen because of accessibility of welding at the mid-wall dimensions of each grid box, and the consequent greater ease in making full penetration welds. Another reason for the choice was the good physical properties obtained in extruded stainless steel shapes. Figure 6–9 shows the cruciform sections which were welded together successively in pairs, rows, and quadrants, and finally into the complete assembly. The normal weld shrinkages of the austenitic series (AISI type-304 and 347) stainless steels, which comprised all the materials in components of the core cage, are much higher than those of carbon steels and therefore presented greater problems in dimensional stability. Cumulative dimensional errors could not be tolerated across the assembly, and elaborate fixturing was therefore set up to build a predicted shrinkage into each weld joint. Small errors were corrected from joint to joint, so that in the over-all weldment few errors exceeded 1/16 in., and adequate machining stock was provided. Final heat treatment did not appreciably affect dimensions. However, it is significant

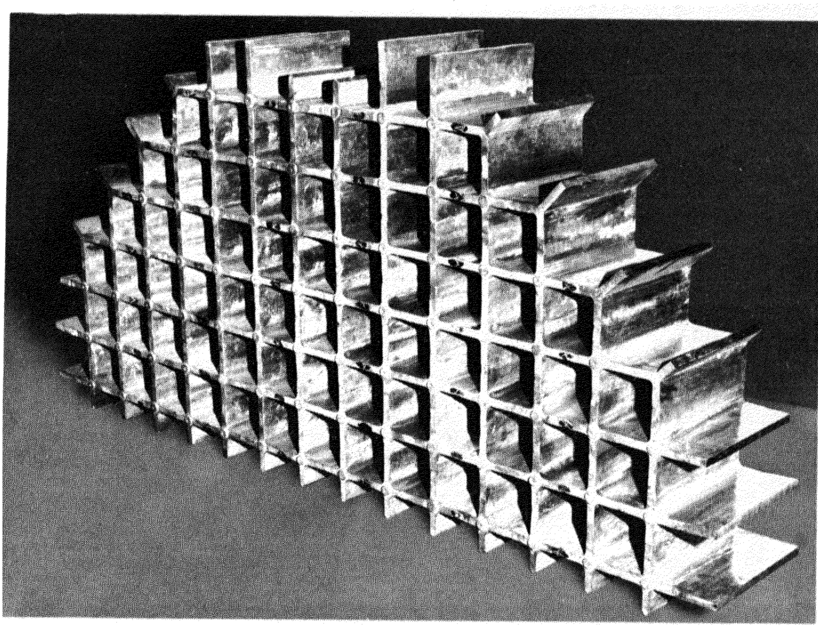

FIG. 6–9. Top grid weldment, half completed.

that these structures expanded more than 1 in. diametrically in rising to the stress relieving temperature (1775°F), and returned to dimension on cooling. This required return to dimension was produced only by rigid control of heating and cooling.

The top grid was machined to a total tolerance, from theoretical box centerlines to box walls, of 0.003 in. The as-welded walls were 3/4 in. thick, but the final machined thickness was 9/32 in. Rough milling reduced the walls close to final dimension, and broaching gave the final finished size. Needed accuracy was maintained by sequential machining. The top grid is shown inside the core barrel in Fig. 6–10. The bottom support plate was similarly constructed insofar as the grids were concerned, but plate covers were welded on top of each individual grid box. The final form gave the appearance of a solid plate. The weldment, as expected, bowed somewhat because of its unbalanced construction, but not beyond acceptable dimensions. A high degree of accuracy was required in this assembly because there was no further machining of the grid walls. Continuous and close inspection and fabrication follow-up were demanded by this work.

Because of early uncertainties in fabricating the bottom support plate and top grid, a parallel effort was initiated to process solid disc forgings, from which these components would be machined. However, in the early

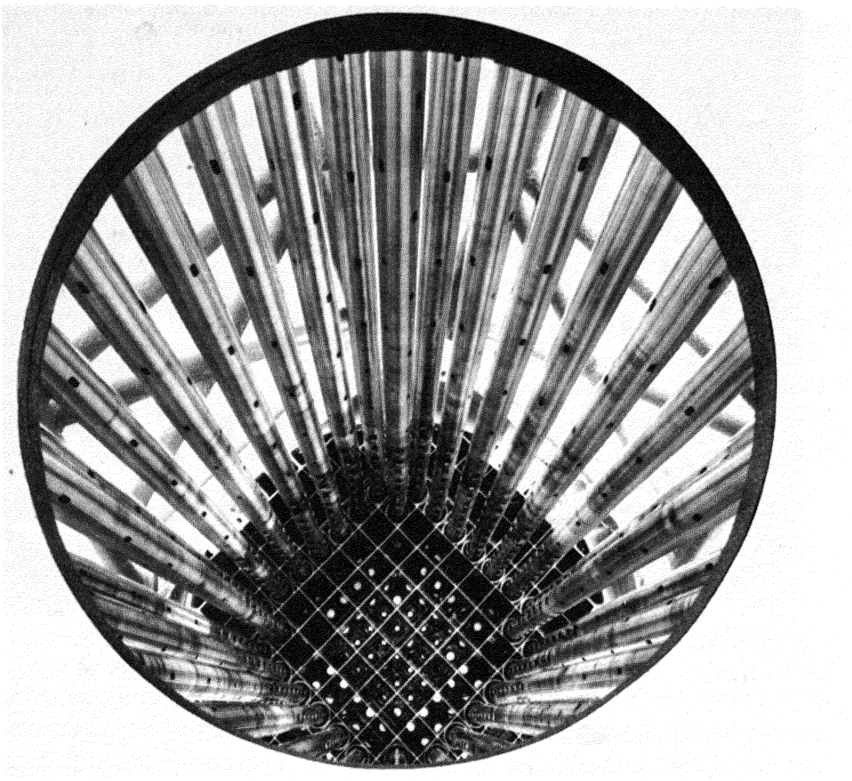

FIG. 6–10. Top grid in place, showing alignment of control rod shrouds and seed fuel assemblies.

stages of machining the weldments, it was apparent that they were reasonably sure of success, and the forgings were abandoned. The thickness and girth of the forged discs made them among the largest stainless steel forgings ever produced. In fact, all components of the core cage taxed the facilities of basic steel producers.

6–4.3 The core barrel and the hold-down barrel. The core barrel and the hold-down barrel were produced from 2-in., 3-in., and 4-in. plates. Heavy ring forgings were the raw stock for the flanged ends. The fabricating sequence for the core barrel consisted of forming the barrel sections, welding the required girth seams, and machining to a tolerance range of 0.020 in. on a $92\frac{3}{4}$ in. OD over the full 155-in. length. Stress relieving at 1775°F for 10 hr was required for dimensional stability during the final machining. All machining was performed on a vertical boring mill. The flimsiness of such a structure is demonstrated by the fact that it would go out of tolerance by virtue of its own weight if laid on its side unsupported.

Internal supports were required in both handling and shipping to maintain true diameters.

Fabrication of the hold-down barrel was similar in nature to that of the core barrel.

6–4.4 The control rod linkages and shrouds. Each control rod is linked to its activating mechanisms by a linkage consisting of a hollow drive shaft and an internal tie rod. The over-all length of the assembled unit is 261 in. and the outside diameter is $1\frac{3}{8}$ in. High tensile strength, wear resistance, and impact properties are required of this unit. Both the drive shaft and the tie rod are made from an age hardenable stainless steel (17–4PH) having 17% chromium, 4% nickel, and 3% copper as principal elements. The control rods must be protected from lateral hydraulic loads in the core; they are guided through bearing arrangements by the control rod shrouds — 17-ft long by 4-in. diameter rigid pipe assemblies containing bearing blocks at intervals. These slender shrouds presented alignment problems during welding. The premachined components received no further machining after welding. The high functional integrity required in the shroud and control rod assemblies was produced without major difficulty.

6–5. Instrumentation

6–5.1 Terminology and joining techniques. The three types of instrumentation used in the core are (1) *thermocouples* positioned to measure temperatures in appropriate locations throughout the core; (2) *flow measurement instrumentation* (FMI), having flow nozzles or modified venturis for measuring flow in seed clusters and blanket assemblies; and (3) the *failed element detection and location instrumentation* (FEDAL), having sampling rakes on blanket assemblies to draw off exit coolant samples which are transmitted to external fission product monitors. Temperature measurements permit evaluation of flux tilt, power sharing between seed and blanket, power distribution across the core, and axial flux or power distribution. Flow measurement permits evaluation of the flow distribution in the seed, and the relative flow in each blanket region.

The joining techniques for all instrumentation components involved brazing and welding. Brazing of joints in contact with reactor water was done with Nicro-Braze, a nickel-chromium-boron alloy having excellent corrosion resistance. Brazing was accomplished in a hydrogen atmosphere furnace at 2100°F and held at temperature for periods up to 20 minutes. Welding was done principally by the shielded inert gas non-consumable electrode method, using AISI type-308 stainless steel as filler. The welding current used was dependent on the mass of material to be joined and was selected on the basis of results obtained from test mockups.

6–5.2 Temperature sensing instruments. Temperature sensing is achieved by means of 114 thermocouples having AISI type-304 stainless steel sheaths and zirconium oxide insulation. Since these sheathed thermocouples are exposed to water at 2000 psi reactor pressure, unimpeachable integrity of the sheath is necessary. This integrity was checked by means of dye penetrant or fluorescent penetrant tests and by helium leak tests in which the sheath and welded hot junctions were subjected to helium under high pressure. Leaking helium was detected by means of a mass spectrometer.

The temperature sensing ends of the thermocouples are affixed in the proper positions by mechanical means, and the leads are brought up through the core and the reactor vessel in stainless steel tubing or in braided stainless steel conduit. Their exit from the vessel is made through instrumentation ports, bosses on mechanism housings, and terminal boxes. The primary seal between the reactor water at 2000 psi and the outside atmosphere was effected by brazing the thermocouples into a plug by means of Nicro-Braze. The plug then became part of a hermetically sealed assembly at the reactor vessel head.

6–5.3 Flow measurement instruments (FMI). The flow in 16 of the seed clusters and 20 of the blanket assemblies is measured by means of flow nozzles and modified venturis. The hydraulic responses used to determine flow rates are transmitted through about 30 ft of 1/4 in. OD type-304 stainless steel tubing which runs sinuously around the working portions and structural supports in the reactor and emerges through instrumentation ports in the vessel head to connect with the indicating or recording instruments.

Two tubes are required at each of the 36 positions where flow is measured. Fabrication of the tubing involved much precision bending and joining. All bends were made to a template and checked optically to maintain close tolerances. All tubing runs terminated in tube sheets to which the tubes were joined by brazing with Nicro-Braze. The lower portion of the FMI System extends downward from the top grid to the bottom support plate, where it extends outward to the individual positions of the flow sensing devices. This portion was assembled by brazing groups of six tubes into tube sheets, the tubes being sheathed by a conduit threaded to the tube sheet. This assembly was then bent into a predetermined configuration following the contour of the barrel and extending to a point 1 in. above the bottom support plate. From this point a square conduit (formed by welding two channels together) encases the tubes to a point where they are welded into the bottom support plate adapters. Both welding operations were performed using the shielded inert gas nonconsumable electrode method.

The upper portion of the FMI System extends upward from the top grid and consists of three conduits, each having 24 $\frac{1}{4}$-in. OD tubes, which rise vertically and leave the vessel head through three instrumentation ports. The upper portion is connected to the lower portion through a short connector with mechanical fittings. The integrity of the system was evaluated by helium leak testing.

6–5.4 Failed element detection and location (FEDAL) instruments. The exit water from each of the 113 blanket assemblies is sampled and monitored for fission products. The water is piped from the sampling rake contained in each blanket assembly to the monitor outside the reactor through about 40 ft of 5/16-in. OD tubing of type-304 stainless steel.

The water from each sampling rake descends through the sampling tube inserted in the corner of the blanket assembly; travels through a bellows type flexible connector assembly, the bottom support plate adapter, and the horizontal tubing running across the bottom support plate; ascends through the riser along the wall of the core barrel; then flows through the junction box and out of the reactor to the monitor.

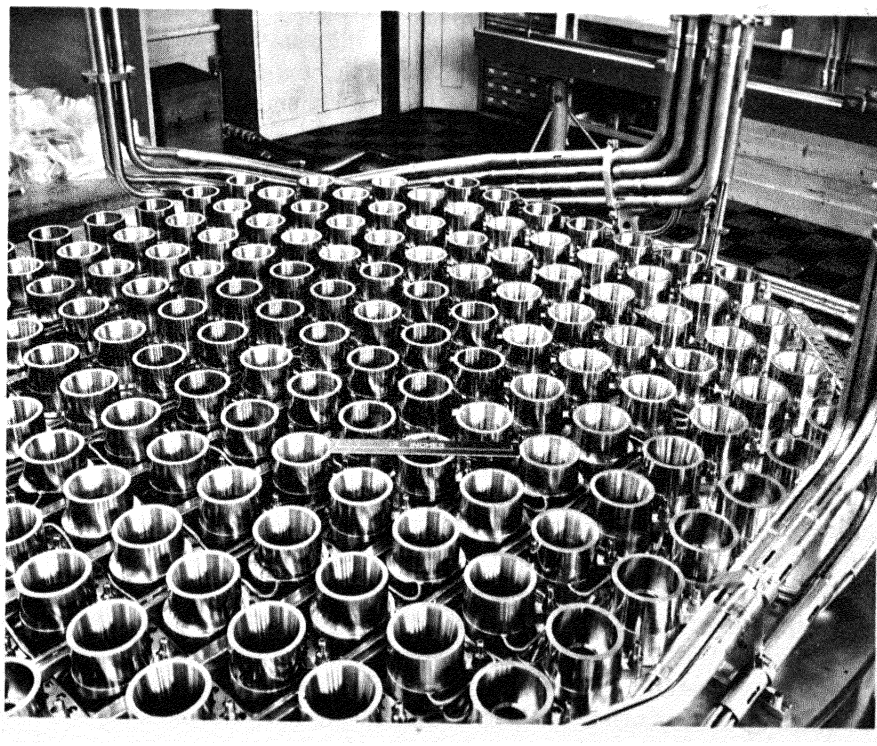

FIG. 6–11. Bottom support plate with adapters in place, showing instrumentation.

The individual tubes which run sinuously over the bottom plate to terminate at the adapters as shown in Fig. 6–11 are collected into groups of seven tubes each to make the rise from the bottom support plate to the top grid. The seven tubes were assembled by grouping six tubes symmetrically around a seventh central tube which had several spacers welded to it. These seven tubes were brazed into a tube sheet. This assembly was then sheathed with a slotted conduit welded to the tube sheet. All bends of this assembly were made in the 5/16-in. tubes as a group. The bends were then encased in conduit fabricated as split elbows and joined by welding. The elbows and straight conduits were joined by sleeves with welding at both elbows and straight conduits. All welding was done using the shielded inert gas non-consumable electrode method.

At the top grid, four groups of seven tubes were collected together to continue the vertical rise to the junction box. At the junction box, four of the larger groups (including all 113 tubes) were collected together for the final vertical rise out of the reactor vessel head through a single instrumentation port.

The watertight joints at the top grid and in the junction box were made with gold seals at the interfaces between mating tube sheets. A ball-and-cone type joint was used to make a watertight seal between the sampling tube connectors and the spring-loaded flexible connector assemblies. The same type of joint was used between the flexible connector assemblies and the bottom support plate adapters. All bends, as in the FMI System. were made to a template and checked optically.

6–6. Assembly Operations and Tests

Components of the Shippingport reactor were assembled at the point of manufacture. Involved were fuel unit assembly; instrumentation fabrication; assembly of the core cage; and trial fitting of control rod assemblies, shroud assemblies, and fuel unit assemblies into the core cage. After all checks were made, components were shipped separately to the site. Here all components were assembled into a completed core, ready for insertion into the pressure vessel. The above items, along with core assembly tooling, are covered in more detail in the remainder of this section.

6–6.1 Assembly tools. The design, manufacture, and operation of assembly tools required care equal to similar operations on core components themselves. Wherever possible, tools were designed for multiple assembly functions. Assembly tools encompassed a wide range of size, weight, and complexity. Limited space for tool operation, complex geometry of core components, and the difficult accessibility of core components during and after assembly were factors necessitating numerous special tools. Cleanliness requirements for all tools were the same as for core components.

Core assembly tools fell into three general categories: (1) handling tools, (2) jigs, and (3) fixtures. In general, handling tools served for lifting, storing, and shipping, with some used for more than one of these functions. Jigs were positioning tools employed for precise work such as an optical alignment or a measurement requiring an accuracy of 0.005 to 0.010 in. or less at distances of from 1 to 30 ft. Fixtures were guiding and/or supporting tools required for specialized assembly operations.

6–6.2 Fuel unit assembly. Fuel units were largely assembled at the point of manufacture. The operation consisted chiefly of assembling with special tools the seed and blanket Zircaloy components with the stainless steel and Inconel parts. Thermocouples were also installed in six special instrumented seed assemblies. A mockup was made to represent the maximum misalignment between the bottom support plate and the top grid that might be expected in the completed core. All seed and blanket assemblies were installed and latched into this mockup to check for proper performance. Prior to shipment to the site, all seed assemblies were also installed in their proper locations in the core cage and a few blanket assemblies were latched into the core cage at critical locations to provide final assembly installation at the site without interference from instrumentation conduits.

All fuel units were removed from the core cage and shipped as individual components. At the site, seed and blanket assemblies were finally installed and latched in the core cage.

6–6.3 Instrumentation assembly. Flow measurement instrumentation (FMI) and FEDAL instrumentation below the top grid were fabricated and installed in the core cage at initial assembly. By employing special tooling, such as templates for bending instrumentation tubing and fixtures to support and align the instrumentation conduits, the instrumentation below the grid was installed on the bottom support plate before inserting it into the core barrel. At this stage, five inlet water thermocouples were installed in the bottom support plate.

Instrumentation above the grid was fabricated and shipped for final assembly without prior fitting to the core cage. At final assembly, the hold-down barrel was installed on top of the core cage, and by the use of a special fixture simulating the instrumentation exit through the pressure vessel head, the above-the-grid FMI and FEDAL instrumentation was installed. Individual connectors for the pressure-tight joint between instrumentation above the grid and instrumentation below the grid were installed and hydrostatically leak-tested before each instrumentation conduit section was inserted into the core cage. Completed instrumentation from the bottom support plate through the upper conduits was hydrostatically leak-tested at 30 psi to ensure structural integrity.

Blanket exit water instrumentation and seed exit water instrumentation were assembled at Bettis and were trial-fitted into a mockup section of the core simulating the clearances and alignments from the pressure vessel head through the top grid. After checking out satisfactorily in the mock-up, all BEWI and SEWI instrumentation was shipped to the site and installed without difficulty in the pressure vessel.

6–6.4 Core cage assembly. For initial assembly, the core barrel was hung in an upright vertical position in a stand, supported under its flange by a levelling ring and eight equally spaced jacks. The top grid was placed under the barrel, raised inside the barrel by a four-legged sling, and held by the sling until forty $1\frac{1}{2}$-in. retaining pins were inserted to retain the grid in a fixed position. To capture these pins, a back-up plate was welded on the outside of the barrel; the metal arc technique, using AISI type-308 stainless steel electrodes, was employed. After the below-grid instrumented adapters and thermocouples were installed on the bottom support plate, the core barrel and top grid assembly was raised by a 50-ton crane and the instrumented bottom support plate assembly located directly beneath it. The core barrel and top grid assembly was then lowered on the levelling ring and the bottom support plate assembly raised into place and held by a four-legged sling. Then 24 retaining pins were installed and captured by arc-welded backup plates. Soundness of welds was determined by dye penetrant inspection.

Top grid openings and the adapters in the bottom support plate were referenced to the true centerline of the core barrel. They were mutually aligned by means of a special jig bar and optic tooling methods. The five grid seal housings positioning and joining the above-grid and below-grid instrumentation were secured by dowel pins and stay bolts.

When initial assembly and trial fitting of fuel, control rods, and shrouds were completed, the core cage was shipped in a special shock-mounted and pressurized container. At the site, the core cage was supported in another stand by methods similar to those used for initial assembly. The core was now ready for fuel to be inserted and for other final work.

6–6.5 Control rod and shroud assembly. To check core assembly alignment, with all seed fuel assemblies in place, the control rod shrouds were hung from a shroud support plate that simulated the pressure vessel head. The plate was optically aligned above the top grid and was rigidly supported from the top of the core barrel flange. After the 32 shrouds, shown in Fig. 6–10, were in place, an optical alignment check was made between the top of each 19-ft-long shroud and the lowest stellite bearing, which was 8 ft from the lower end. An open reticule was lowered into the bearing and another placed in the top bore of each shroud. Then, by establishing

a vertical line of sight with alignment telescope through the top reticule, any misalignment of the stellite bearings could be ascertained.

Scram shafts were then installed in each shroud and latched into the control rods. As an additional shroud alignment check, each rod was drawn up into the shrouds to a height of 72 in. and then lowered. If the load required to withdraw a rod did not exceed the total weight of a control rod and scram shaft assembly, the assembly was considered satisfactory. As a final check, 30 scram shafts and shrouds were removed, leaving the two most misaligned shrouds in place. The shroud support plate was then offset from the grid by 1/8 in. in two directions and the control rods were again withdrawn and lowered. The absence of excessive load indicated acceptability.

The above operations were part of initial assembly only and were not repeated at final assembly.

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CHAPTER 7

CHEMISTRY

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7-1. INTRODUCTION

Early in the development of the Shippingport reactor it became evident that a large chemistry effort would be required. This need arose largely from two characteristics of the core not encountered in plants previously designed. First, Shippingport was a large plant with a core containing a large number of fuel elements with high uranium content. This introduced a host of problems related to the possibility that fuel elements might fail and release fission products to the plant. Second, PWR was a developmental plant with removable fuel subassemblies that, on indication of malfunction, could be taken out and examined. This, of course, required the development of a technique for locating failed fuel elements in an operating plant.

The changes from previous designs in the thermal, hydraulic, and mechanical characteristics of the core required, in addition, the re-evaluation of two problems considered in previous work: plant contamination by induced radioactivity and fouling of core surfaces by transported system corrosion products. These studies led to new demand on the coolant chemistry, and these in turn had to be evaluated with respect to the major problem of fission-product release.

Pressurized water reactors are essentially closed cycle plants, and in their conventional chemistry aspects they most closely resemble high-pressure water process heating systems. After an initial charge of base, they can be maintained at a pH level of 10 to 10.5 by means of a bypass purification system employing a basic mixed-bed ion exchange resin, such as LiOH. Purification by this means is effective, and water dissociation does not appear to be appreciably different from operation at neutral pH. The Shippingport plant is operated at pH 9.5 to 10.5 with a LiOH mixed-bed resin.

The origin and solution of various chemical problems dealt with in PWR are best considered in terms of the following three basic functional requirements of the plant:

(1) Assurance of the structural integrity and operability of the plant, including its mechanical and heat transfer components.

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(2) Assurance of continuous and convenient habitability and accessibility (from the standpoint of radioactivity) for operation and maintenance.

(3) Provision for control and disposal of plant wastes in accordance with established standards.

7-2. Structural Integrity and Operability

The structural integrity and operability of the reactor plant are affected by chemical variables determined by: (1) materials used in the reactor and the reactor coolant systems (those systems that handle reactor coolant at operating temperature and pressure), (2) chemistry of the coolant, and (3) specific effects of irradiation, such as deposition of corrosion products on core surfaces. The materials employed and the chemistry of the environment must together meet the requirement for structural integrity and operability as applied separately to reactor coolant system pressure boundaries, the core, and the mechanical components operating in the coolant systems. The specific problems of selecting materials for reactor core cladding and for certain mechanical components in the reactor coolant systems are discussed in Chapters 5 and 12, respectively; here it is necessary to note only that the materials selected on the basis of other considerations are satisfactory for use in neutral or alkaline water having a low oxygen content.

7-2.1 Materials. The two requirements for reactor coolant system pressure boundary materials are adequate resistance to or allowance for general corrosion in the coolant, and the absence of local attack, such as pitting or cracking. When the coolant loop material was selected, adequate experience was available only for stainless steel. In low-oxygen neutral or alkaline water, stainless steel is, by ordinary standards, very corrosion resistant. At 500°F, the corrosion rate is of the order of 5 to 10 mg per square decimeter per month, expressed as Fe_3O_4, representing a penetration rate of less than 0.1 mil per year. Corrosion is uniform, and the corrosion products are primarily insoluble. In neutral water the corrosion solids are found largely in the form of a loosely adherent, finely divided magnetite containing chromium and nickel oxides, and as a thin film on the metal. At high pH, the quantity of loose, solid corrosion products, known as "crud," is considerably reduced and the corrosion film is thicker. The crud is partially transportable through the systems; it is of great significance because of radioactivity induced by exposure to neutrons during transport through, or residence in, the core.

Because of the desire to use high pH coolant water (see Article 7-2.3), the question arose whether there might be a possibility of caustic stress

corrosion cracking if stainless steel were used as the pressure boundary material. It was felt that such cracking might occur when stressed parts were located in areas where caustic substances might concentrate. A survey was made of the plant to identify the possibility of such concentrations and a test program was conducted to determine the cracking tendencies of certain alkali solutions. These indicated that no problem of this nature existed.

7-2.2 Radiation chemical reactions. Oxygen content and pH, which are the most significant properties of the reactor coolant, are seriously affected by two types of radiation-induced chemical reactions that occur in pressurized water reactors. Information on this problem was already available from previous work. First, water can dissociate into hydrogen and oxygen, or these can recombine to form water, depending on the relative concentrations of the reactants. Second, nitrogen can form either ammonia or nitric acid, or either of the latter two can be converted to the other, depending on whether free hydrogen or free oxygen is present in the water simultaneously with nitrogen. These radiation-induced reactions can be summarized as follows:

(1) Water dissociation and formation,

$$2H_2O \rightleftarrows 2H_2 + O_2.$$

(2) Formation of ammonia,

$$3H_2 + N_2 \rightleftarrows 2NH_3.$$

(3) Formation of nitric acid,

$$2N_2 + 5O_2 + 2H_2O \rightarrow 4HNO_3.$$

(4) Reduction of nitric acid to ammonia,

$$2HNO_3 + 5H_2 \rightarrow N_2 + 6H_2O.$$
$$N_2 + 3H_2 \rightarrow 2NH_3.$$

(5) Conversion of ammonia to nitric acid,

$$4NH_3 + 3O_2 \rightarrow 2N_2 + 6H_2O.$$
$$2N_2 + 5O_2 + 2H_2O \rightarrow 4HNO_3.$$

In pressurized water reactors at 500°F, the hydrogen-oxygen equilibrium is such that oxygen is not detected if the hydrogen concentration is above 5 cm^3/kg. The kinetics of the reaction in either direction are quite rapid with the reactor operating. Thus, if an excess of hydrogen is maintained in the coolant, oxygen trapped in or added to the system can be rapidly converted to water.

Although at full reactor power the hydrogen-oxygen recombination rates are too rapid for convenient observation, the ammonia synthesis and decomposition are much slower. In the range of hydrogen concentration studied, the forward reaction does not appear to be dependent on the hydrogen concentration. The nitric acid reactions have not been observed

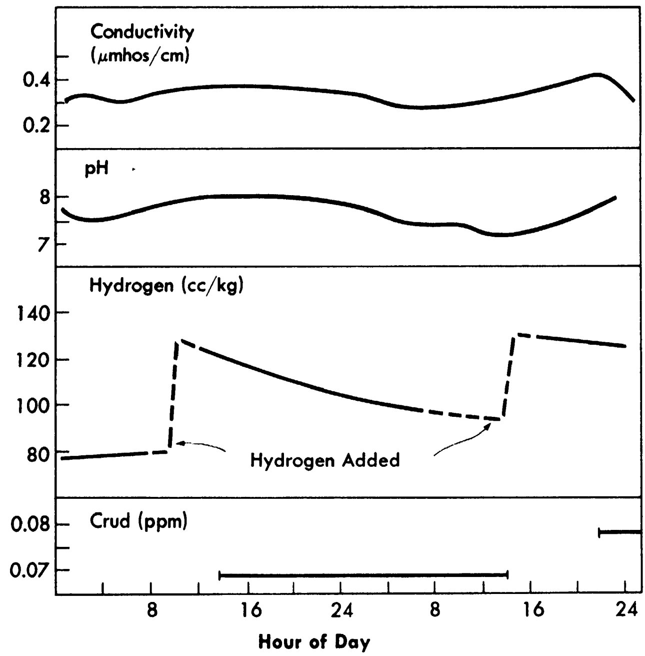

FIG. 7–1. Water chemistry; typical operating log with temperature at 450°F, filter and demineralizer in service.

quantitatively in operating plants because of the potential damage to plant materials, but they appear to be more rapid than the ammonia reactions and slower than the hydrogen-oxygen reaction.

The effect of these reactions on the chemistry of pressurized water reactors is illustrated in Figs. 7–1 and 7–2. Figure 7–1 shows the chemical conditions with high hydrogen and some nitrogen in the water. The conductivity and pH of the water are largely the result of ammonia, the concentration of which is determined by the rate of synthesis and the rate of removal by a mixed-bed ion exchange purification system. Hydrogen losses are made up by additions. The concentration of suspended solids, or crud, is quite low. Figure 7–2 shows experimental results with high oxygen and nitrogen in the system, in the form of air added at startup. As the power level and temperature were increased, nitric acid formed and neutralized the ammonia present from previous operation. Conductivity and pH decreased until pH 7 was reached; when the water became acid the conductivity showed a corresponding increase. With the formation of nitrate, a corresponding increase in chromate ion was observed, resulting from chemical attack on the materials. The whole process was readily reversed by adding hydrogen, which combined with the oxygen and reduced the nitrate and chromate ions.

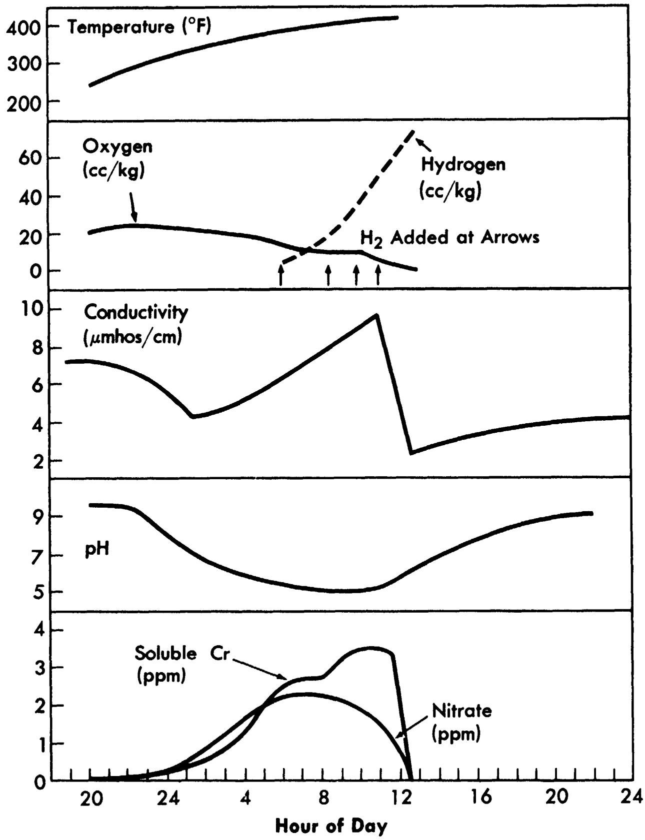

FIG. 7–2. Water chemistry; experimental startup with air in loop.

These results, plus the requirements cited earlier, indicate the general undesirability of oxygen in such plants. Accordingly, oxygen is removed from the water either chemically with hydrazine at about 200°F if the reactor is not operating, or by combination with hydrogen in reactor flux, and hydrogen is maintained in the water at all times. In the Shippingport installation the hydrogen level is maintained at 25 to 50 cm^3/kg. Because deaerated make-up water is used, the only significant nitrogen in the plant is that trapped at filling.

7–2.3 Deposition of solids. The high heat-transfer coefficients employed in pressurized water reactors require that fouling on the fuel elements be significantly lower than in conventional heat-transfer processes. The formation of heavy deposits in operating reactors would be of considerable concern. During the irradiation of fuel elements in in-pile loops, fouling has been observed at times in the form of more or less uniform deposits of

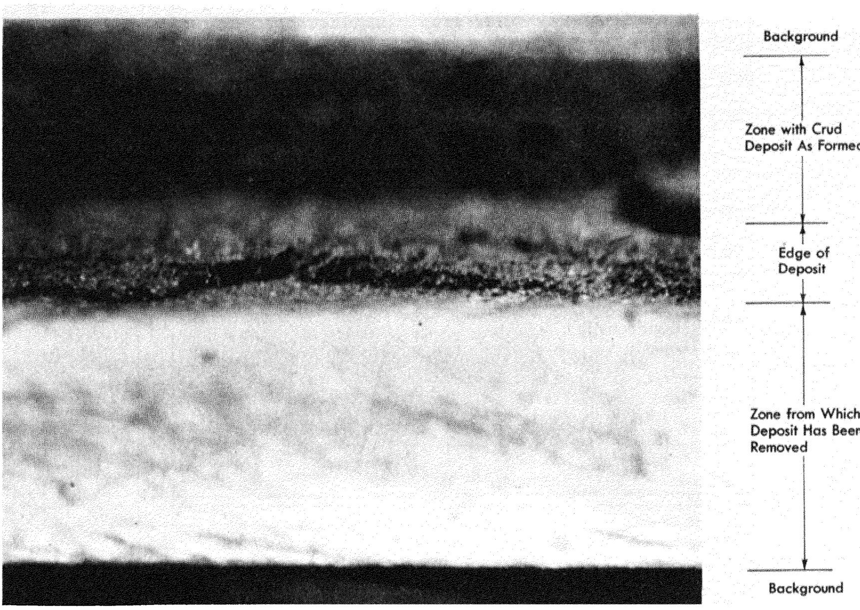

FIG. 7–3. View of deposit on irradiated fuel sample (10×). The photograph was taken through a hot cell window.

finely divided magnetite, a corrosion product of the stainless steel of the loop. Figure 7–3 shows a deposit formed on a fuel sample during irradiation in a high temperature in-pile loop.

The factors contributing to the formation of such deposits have not been completely explained, but some qualitative observations can be made: (1) Deposition to the extent observed appears to be a specific effect of irradiation; the amount of deposit is proportional to the radiation flux. Deposits have been formed at high ionization densities with electron beam irradiation. (2) As first demonstrated in electron beam tests and later in in-pile tests, operation at high pH, above 9.5, produces less deposition than operation at neutral pH. Alkalinity provided by the small amounts of ammonia present in operating plants is beneficial in this respect. (3) The worst case of deposition occurred during in-pile tests at oxygen levels up to 0.3 ppm, significantly higher than those maintained in other tests or in reactor operations. It has also been shown that fouling of out-of-pile heat-transfer surfaces is similarly affected by oxygen, but at much higher oxygen levels. (4) The ratio of corroding surface to the surface on which deposition occurs is considerably higher in in-pile loops than in operating reactors. This is most significant in determining deposition levels where the collection efficiency on the irradiated surface is very high. (5) Deposition appears to be lower at high velocities, but velocities up to 20 fps have not prevented it.

In view of the uncertainties involved as a result of these phenomena, a detailed review was made of the possible effects of such deposition on heat transfer and pressure drop in the PWR. The conclusion was that it would be desirable to raise the pH of the coolant to minimize such fouling. The applicability of high pH operation with LiOH (pH 9.5 to 10.5), together with associated problems, was evaluated. The conclusion was that such operation was feasible. Therefore a pH range of 9.5 to 10.5, attained by the use of LiOH, was chosen as the reference chemical condition for the reactor coolant.

7-3. Habitability and Accessibility for Operation and Maintenance

7-3.1 Description of problem. In an ideal power reactor system of the pressurized water type, all radioactivity except that induced in pure water would be confined to the reactor, properly shielded within its own container. This ideal condition is not realized in practice for a number of reasons: Structural and other materials exposed to the neutron flux in the reactor become radioactive and, by corrosion, release radioactive species to the coolant. Impurities introduced in make-up water (such as sodium and argon) and soluble and insoluble corrosion products are circulated by the coolant through the reactor where they may become radioactive by neutron capture. Solid corrosion products may be deposited on the reactor surfaces, where they are exposed to high neutron fluxes, and the radioactive materials thus formed can reach all surfaces that are in contact with the circulating coolant. In addition, if fuel elements fail, the irradiated fuel and released fission products may be similarly transported and accumulated in the systems. If the reactor coolant systems should leak, activity can escape to the reactor plant container or the main steam system. Accumulated radioactivity also determines accessibility to the shutdown reactor plant and is one of the major water technology problems in the pressurized water reactor.

In any design, the magnitude of this effect must be kept to acceptable levels, and specific areas of high activity must be quantitatively evaluated so that adequate shielding can be provided. The factors determining the magnitude of system contamination are inherent in the core design, the selection of materials, the coolant chemistry, and the provisions for coolant purification. The relationships between these elements are complex.

Previous experience had provided some basis for evaluating the problems of induced activity. However, a major new problem in PWR was that of fission products. Most experimental and development effort in PWR chemistry was in this area.

7-3.2 Induced activities. The problem of induced activity in pressurized water reactors is largely one of transported corrosion products, which may

TABLE 7-1

CALCULATED CRUD ACTIVITY FROM CORROSION OF CORE MATERIALS

Deposited specific gamma activity in Mev/cm^2·sec					
Nuclide	1 month	2 months	6 months	1 year	2 years
Cr-51	57.8	120	309	330	330
Mn-56	105	105	105	105	105
Fe-59	56.4	169	549	702	722
Co-58	40.4	135	600	975	1120
Co-60	384	1520	13,300	50,600	186,000
Hf-175	191	630	2,770	4,480	5060
Hf-181	1180	3540	11,600	14,700	15,100
Total	2014	6220	29,230	71,890	208,440

Note: As a rough average, a deposited gamma activity of 100,000 Mev/cm^2·sec will give a radiation level of about 100 mr/hr near the surface of large components.

originate from materials either in the reactor coolant system (deposited in the core, there activated, and again released) or in the reactor proper. The significant isotopes are Co^{60}, Co^{58}, and Fe^{59} from the stainless steel, and Hf^{175} and Hf^{181} from the control rods. If new materials are introduced, these too must be taken into account.

Although a rigorous method of analysis has not yet been developed, the problem of core material corrosion and wear products can be treated conservatively by fairly simple methods. The specific activities of all core materials are computed as a function of time, and half of the material released by corrosion is assumed to be distributed throughout the reactor coolant systems. Such an analysis was made for Core I. The results are summarized in Table 7-1. The major source of the cobalt activity shown is the stellite used in the core structure.

As stated above, the calculations were made on the assumption that the transported corrosion products are uniformly distributed throughout the systems. Actually, crud and solid-fuel corrosion products tend to accumulate at tees, elbows, and points of low velocity. A study of the deposition and settling characteristics of crud and UO_2 indicated the advisability of minimizing traps in the system. Accordingly, the design of the reactor coolant systems was reviewed to minimize such traps, and note taken of the most likely points of collection in the systems as built, so that they could be monitored for activity buildup during plant operation.

7-3.3 Fission product release. In the present state of development, economical nuclear power from pressurized water reactors requires the use, in whole or in part, of fuel elements high in natural or slightly enriched uranium. As alloys, such fuels have high corrosion rates, or if they are in the form of stable oxides, the fuels readily release the more volatile and. soluble fission products. To protect the fuel material and prevent fission product release, the fuel is enclosed in corrosion-resistant cladding. However, it cannot be expected that this cladding will remain perfect over the life of the fuel element. Since the cladding is the sole line of defense against release of fission products to the reactor coolant, some provisions for fuel-element failure must be included in the design of the reactor.

The number of cladding defects that will be experienced cannot now be predicted, but considerable design and fabrication effort has been made to prevent them. The number of failures might be quite small or even nonexistent; however, for design purposes it had to be assumed that some defects will occur in the approximately 100,000 fuel elements in the reactor.

Four steps had to be taken: (1) the possibility of gross fuel element failures had to be determined; it had to be demonstrated that defects in individual fuel elements of the chosen design would not lead to progressive failure of neighboring elements and thus of substantial parts of the core. (2) Data had to be obtained on the release of fission products from defective fuel elements, by irradiation tests on fuel elements deliberately made defective. This was necessary to establish a basis for the design of the purification and waste disposal systems so that the plant could operate, if necessary, despite these defects over the full useful lifetime of the core. (3) A means had to be developed to locate failed fuel elements so that these could be removed for study and, in the case of major failures, to limit activity in the plant. (4) Methods had to be developed to decontaminate the reactor in the event of a major failure that would prevent access to the plant for maintenance. These aspects of the problem are considered briefly below.

Possibility of gross fuel element failures. (a) Seed fuel elements. Pure uranium metal and high uranium content alloys exposed to high temperature water will in most cases corrode rapidly to form UO_2 particles; the UO_2 product has a larger volume than the original metal. Thus, water entering a defect in an element would cause it to burst, allowing the UO_2 corrosion particles with their associated radioactive fission products to be released and dispersed throughout the coolant system. This would seriously contaminate the plant. It also might obstruct coolant flow passages and cause overheating and progressive failure of adjacent elements. The seed fuel in PWR avoids these consequences by utilizing a low content of enriched uranium alloyed with zirconium. The alloy has good corrosion

resistance, and a defect in the cladding would be unlikely to produce gross failure or significant contamination problems during normal operation of the plant.

(b) Blanket fuel elements. The nuclear concept of the seed-blanket core in PWR requires the use of material with high uranium content for the blanket fuel elements. If uranium metal or high uranium content alloys were used, the problems stated above would again hold. It was largely this consideration that governed selection of UO_2 as the blanket fuel for PWR. UO_2 is stable in high temperature water; thus a defective UO_2 rod element would not rupture and release particles as might a high uranium content metal fuel. Tests were nevertheless made to determine if water would enter a defect, form steam, and burst the UO_2 element to cause coolant passage obstruction and possibly progressive failure of adjacent elements. These tests showed that, even under grossly exaggerated conditions, progressive failure did not occur. The nature and amounts of fission products released were studied, and the results are discussed in the following paragraphs.

Fission-product release from defective UO_2 fuel elements. From experience in the fuel element manufacturing and irradiation programs, the following assumptions were made with respect to the possible incidence of defects in the PWR UO_2 rods:

(1) Not more than 1000 fuel elements in the PWR blanket would be likely to develop defects.

(2) Of these, not more than three percent would be likely to develop significant cladding failures.

From the manufacturing process it was concluded that the defect most likely to occur in a UO_2 rod was a small hole from weld porosity or weld contamination. Accordingly, irradiation tests were made on fuel elements with small holes drilled through the cladding to represent such defects.

TABLE 7–2

ESCAPE RATE COEFFICIENTS, ν, FOR A NORMAL PWR FUEL ELEMENT

Relative order	Elements	ν, sec^{-1}
1	Cs, I, Xe, Kr, Rb, Br	1.3×10^{-8}
2	Mo	2×10^{-9}
3	Te	1×10^{-9}
4	Sr, Ba	1×10^{-11}
5	Zr, Ce, other rare earths	1.6×10^{-12}

Tests were also made with cracks, and with multiple defects per rod, under steady and cycling conditions. The effect on UO_2 melting of fuel clearance in the rod was developed, and the relationship of these conditions to fission-product release determined. A final test of one year duration was set up with a reference UO_2 rod to determine the effect of burnup and exposure on fission-product release.

Results of these experiments are summarized in Table 7-2, which presents the escape rate coefficients ν for fission products from a reference UO_2 fuel rod.* The escape rate coefficients of the elements appear to vary relatively in the same way as the volatility and solubility of the chemical species. They are highest for elements in Groups I, VI, VII, and VIII of the periodic table and are lower for Groups II, III, IV.

Absorption characteristics and resulting contamination follow the opposite trend. They are essentially zero for Groups V, VII, and VIII, and highest for Groups II and IV. Tellurium, Group VI, is also strongly absorbed. It should be noted that to be significant for contamination, an isotope must have an appreciable half-life and be an energetic gamma emitter. These and the nature of the decay chains combine to favor the use of UO_2 as a fuel, as illustrated by consideration of three significant contaminant isotopes: Zr^{95}, Y^{91}, and C^{141}. The isotope Zr^{95} has very short half-lived precursors and a low escape rate coefficient ν and therefore, by both indirect and direct paths, has a low escape to the system. Similarly, Y^{91} has short-lived Kr and Rb precursors, and although its immediate precursor Sr^{91} has a significant half-life (9.7 hr), the escape coefficient of the Sr is low. The considerations for Ce^{141} are analogous to those of Y^{91}. A similar situation applies to other isotopes that are significant radiological hazards, for example I^{131} and Sr^{90}. Although I^{131} escapes readily, it is not absorbed in the systems and is readily removed by an ion exchanger.

With the information thus obtained on the release of fission products, analytical studies could be made of the fission products which could be expected in the reactor for a given number of defects, and of the requirements and expected performance of the purification and waste disposal systems. Table 7-3 gives the fission-product activities calculated from the data of Table 7-2.

These studies indicated that system degasification would not be useful for control of fission products in PWR (because the gases have short

* Fission products escape from UO_2 by two mechanisms: recoil of fission products from the surface of the fuel material, and diffusion of accumulated fission products. The first mechanism is most significant for short-lived activities; the latter is more significant for the longer-lived activities which can accumulate to substantial levels. Results of these measurements on longer-lived activities are expressed as escape rate coefficients, ν, which are the fraction of accumulated fission products escaping per unit time.

TABLE 7–3

ESTIMATES OF FISSION PRODUCT ACTIVITY IN THE PWR REACTOR COOLANT AFTER 3000 HOURS OF STEADY FULL-POWER OPERATION WITH ONE DEFECTED BLANKET FUEL ROD

Isotope or Chain	Fission yield, %	Parent, $\lambda_i N_{w_i}$, dis/sec·cm^3	Daughter, $\lambda_j N_{w_j}$, dis/sec·cm^3
31.8 m Br-84	0.65	0.132	
10.3 y Kr-85	0.273	1.67	
4.36 h Kr-85 m	0.97	2.05	
78 m Kr-87	2.7	1.71	
2.8 h Kr-88 → 17.7 m Rb-88	3.14	4.27	4.11
15.4 m Rb-89 → 54 d Sr-89	4.78	0.527	5.26×10^{-3}
28 y Sr-90 → 64 h Y-90	5.77	4.58×10^{-5}	1.02×10^{-5}
9.7 h Sr-91 → 58 d Y-91	5.07	3.95×10^{-3}	9.57×10^{-4}
2.7 h Sr-92 → 3.6 h Y-92	5.25	2.87×10^{-3}	1.64×10^{-3}
10 h Y-93	6.0	9.45×10^{-4}	
16.5 m Y-94	6.0	1.41×10^{-4}	
65 d Zr-95 → 35 d Nb-95	6.00	1.13×10^{-3}	1.13×10^{-3}
23.3 h Nb-97	5.73	9.94×10^{-4}	
17 h Zr-97 → 72 m Nb-97	6.5	1.10×10^{-3}	1.14×10^{-3}
30 m Nb-98	6.0	2.32×10^{-4}	
67 h Mo-99	6.14	1.12	
14.6 m Mo-101 → 14.3 m Tc-101	5.6	0.118	0.111
11.5 m Mo-102	4.1	6.70×10^{-2}	
40 d Ru-103	2.85	5.34×10^{-4}	
36.5 h Rh-105	1.35	2.36×10^{-4}	
1 y Ru-106 → 30 s Rh-106	0.38	1.69×10^{-5}	1.62×10^{-5}
18 s Rh-108	0.15	7.29×10^{-8}	
30 h Te-131 m → 8.14 d I-131	0.416 (P_i)	0*	2.84*

		3.02 (P_j)	0.367†	2.77†
77 h Te-132	→ 2.4 h I-132	4.49	0*	77.9*
			4.12†	4.18†
20.8 h I-133	→ 5.27 d Xe-133	6.62	5.70	368
44 m Te-134	→ 53 m I-134	8.00	0*	4.30*
			2.23†	3.80†
67 h I-135	→ 9.13 h Xe-135	6.35	4.61	16.7
30 y Cs-137		6.15	4.56×10^{-2}	
17 m Xe-138	→ 33 m Cs-138	5.74	0.796	1.81
9.5 m Cs-139	→ 85 m Ba-139	5.79	0.413	0.242
12.8 d Ba-140	→ 40 h La-140	6.32	5.91×10^{-3}	1.40×10^{-3}
18 m Ba-141	→ 3.7 h La-141 → 33 d Ce-141	5.7	7.20×10^{-4}	$\lambda_j N_{w_j} = 9.55 \times 10^{-4}$
				$\lambda_k N_{w_k} = 1.07 \times 10^{-3}$
33 h Ce-143	→ 13.7 d Pr-143	5.2	9.20×10^{-4}	9.80×10^{-4}
282 d Ce-144	→ 17.5 m Pr-144	5.39	3.11×10^{-4}	2.32×10^{-4}
5.9 h Pr-145		4.4	6.19×10^{-4}	
24.6 m Pr-146		2.93	9.63×10^{-5}	
			dis/sec·cm^3 (1 rod)	Percent gross activity
Total rare gases			395.196	79.15
Total halogens			95.482	19.12
Total alkali metals			6.906	1.38
Total molybdenum			1.305	0.26
Total alkaline earths			0.261	0.05
Total other			0.128	0.02
Gross activity			499.278	99.98

* Tellurium assumed to deposit irreversibly on surfaces. This results in higher iodine activity in water, with no tellurium activity actually in the water.

† Tellurium assumed not to deposit on system surfaces. This results in higher tellurium activity in water.

half-lives) and that a moderate purification rate (80 gpm ion exchange) would control the long-lived soluble fission products. Moreover, the data (1) indicated that the waste disposal requirements could be satisfied by a system providing holdup for decay and demineralization, and (2) provided the basis for evaluating the purification and waste disposal systems.

Failed fuel element detection and location. The core is designed so that fuel assemblies can be removed individually through the reactor vessel head. Full utilization of this feature of the design depends upon instrumentation for locating and measuring fuel element failures. Although much attention had been given to this subject in connection with other reactors, the problem had not been fully explored for a recirculating pressurized water reactor. Analysis showed that for locating defects, any radioactive signal would have to have a short half-life, and the relationship of signal to background would be dependent on half-life. Irradiation tests on a defective fuel element showed that of the gamma and delayed neutron emitters, only the latter had the required characteristics. Accordingly, the delayed neutron counter was investigated, and this method of detection selected for the Shippingport reactor.

On the basis of the tests, it was at first believed that a defect as small as a 5-mil hole could be located. However, it was later discovered that the zirconium used in the Shippingport reactor contains approximately 1.5 ppm of uranium. The background radioactivity from this uranium decreases the sensitivity of location so that only major defects or a large number of small defects in a single channel can be located. This background has a similar adverse effect on all other methods of location studied for this reactor. Effort is being made to reduce the uranium content of zirconium for future PWR and other reactor applications.

7–3.4 Decontamination. By virtue of its design and of the selection of fuel materials for the first core, it is expected that the need for PWR decontamination will arise very infrequently. However, to provide wide latitude for selecting fuel materials and future core designs and to cope with the highly improbable but not impossible failure of metal seed fuel elements, a program to establish suitable decontamination procedures was undertaken. Such procedures had to be effective, practical, convenient, and rapid, with no damage to plant components. Previous experience showed that corrosion by decontamination solutions might present the greatest problem in meeting these criteria.

As indicated previously, the two sources of contamination in reactor plants are induced and fission product activities distributed as solubles and solids. Soluble activities are relatively easy to cope with except for irreversible absorption on solids; the major contamination problem results from activities on or in solids. The basic requirement for elim-

inating activity associated with solids is removal or destruction of the solid by chemical or physical treatment, and the complexing and removal of active species released in the treatment, without harming system components.

The effectiveness of various dissolution-complexing formulations to remove activities in UO_2 and crud was extensively studied as was the corrosiveness of the formulations on stainless steel. Included in this study were oxidizing and reducing agents of low and high concentrations in a pH range of 1.5 to 12. A comparative study was made of the effectiveness of various components of such formulations (oxidants, reductants, complexing agents, stabilizers, inhibitors, surfactants, acids, and bases) for the dissolution of inactive UO_2 and synthetic crud, and their effect on the corrosion of stainless steel. There were other studies of the thermal stabilities of the various components at temperatures of interest. Tests followed with radioactive materials on a bench scale, and finally with decontamination runs in laboratory loops in which decontamination factors of 5 were obtained for UO_2 and crud. A test on a contaminated in-pile test loop resulted in decontamination factors of 30 to 60. No decontamination has yet been attempted on the PWR plant itself.

The results of studies to date may be summarized as follows:

(1) Potentially useful decontamination formulas have been established which, in the fill and flush procedure with moderately high concentrations of reagents or the feed and fleed procedure using very dilute reagents removable by ion exchange, can dissolve as much as 1000 ppm of UO_2 or crud in eight hours of treatment. Formulas now under study are shown in Table 7–4.

(2) Generally, UO_2 is readily soluble in oxidizing-complexing solutions over the pH range of 1.5 to 12. Of numerous oxidants investigated, hydrogen peroxide stabilized with acetanilide was found to be the most effective oxidant under conditions not severely corrosive to stainless steel. Even weakly complexing agents (sulfate, condensated phosphates, etc.) are satisfactory for dissolving UO_2 in very acidic peroxide solutions, although nitriloacetic acid derivates and α-hydroxy acids are generally more effective at high and moderate pH. Somewhat better results are obtained in dissolving UO_2 at low pH than at high pH. Reducing solutions do not dissolve UO_2.

(3) Crud is more difficult to dissolve than is UO_2 and is most effectively dissolved in reducing-complexing solutions at pH 1.5 to 2.0. Increasing the pH reduces the solubilization, while lower pH increases the corrosiveness of the solution on stainless steel. Ethylene-diamine-tetraacetic acid (EDTA) has been found to be the most effective complexing agent, and hydrazine and hypophosphite are the preferred reducing agents for dissolving crud. Oxidizing media under conditions not severely corrosive to

TABLE 7-4

DECONTAMINATION SOLUTIONS
(Concentrations in grams per liter)

Components	Solution					
	Dilute*			Concentrated†		
	Oxidizing		Reducing	Oxidizing		Reducing
	A	B	C	D	E	F
NTA (nitrilotri-acetic acid)	0.3			5		
EDTA (ethylene-diamine-tetra-acetic acid)		0.2	0.3		3.5	3.5
H_2NSO_3H (sulfamic acid)	0.2	0.2	to pH 1.5	~12	~12	~25
Acetanilide	0.25	0.25		1.5	1.5	
H_2O_2	0.3	0.3		3	3	
Hexamine	0.1	0.1	0.1	1	1	1
H_3PO_2			~0.5			
Hydrazine						5
pH	1.5–2.0	1.5–2.0	1.5–2.0	1.5–2.0	1.5–2.0	1.5–2.0
Temperature, °F	275–325	275–325	275–325	200–250	200–250	200–250

* For continuous "feed and bleed" operation using ion exchange cleanup.
† For "fill and flush" batch operation.

stainless steel are far less capable of dissolving crud than reducing media. Crud resists solution by alkaline media, oxidizing or reducing.

(4) The preferred temperature range for dissolving UO_2 and crud was found to be 200 to 300°F. The effectiveness of the solutions and their corrosiveness on stainless steel increase with increasing temperature, but at temperatures considerably above 300°F the thermolysis of the reagents is rapid.

(5) Corrosion of stainless steel, particularly in acidic oxidizing solutions, is markedly reduced by the presence of a suitable inhibitor. Hexamine in the amount of 0.1 to 0.2 g/liter has been found suitable. Acidic decontaminating compositions containing halide ions have been found severely corrosive to stainless steel. At the same pH, decontaminating compositions containing sulfuric acid are more corrosive to stainless steel than corresponding formulations with sulfamic acid, while the latter acid exhibits superior descaling properties. When used as a complexing agent at low pH, α-hydroxycarboxylic acids are less effective in dissolving crud and more corrosive on stainless steel than EDTA.

(6) Wetting agents do not affect dissolving crud and UO_2.

(7) Solutions of the above compositions used under the conditions outlined, that have proved effective for dissolving UO_2 and crud and show promise for decontaminating, dissolve less than 150 mg/dm^2 of 304 stainless steel in a decontamination cycle. These solutions severely attack ordinary carbon steel. If the pH is raised to the extent that the solutions no longer dissolve crud, they are ineffective in decontaminating.

(8) The concentrated solutions given above have been found effective in descaling and, to a degree, in decontaminating crud-coated stainless steel. The pH range for satisfactory descaling without etching is 1.5 to 2.0. Above pH 2.0, descaling is ineffective. At pH 1.25, slight etching occurs. At pH 1.0, etching of stainless steel is severe at the temperatures given for these solutions.

(9) Dilute oxidizing-complexing solutions of the compositions given above can be completely deionized by mixed-bed ion exchange. In experimental loop runs with comparable solution compositions, decontamination of the facility with deposition of the activity in the ion exchange column has been carried out.

(10) Autoclave corrosion studies of the solutions given above with the materials of construction of a pressurized water reactor have shown tentatively that only copper alloys, nitrided surfaces, sensitized 304 stainless steel, stressed 410 stainless steel, and A-106 plain carbon steel suffer more than insignificant attack.

On the basis of these results, procedures have been prepared for the decontamination of the Shippingport reactor, but no decontamination has yet been attempted in the plant. Work is continuing on improvement of

decontamination procedures for high-uranium fuels, and on the development of methods for decontamination of corrosion products from high-zirconium fuels such as PWR seed plates.

7–4. Disposal of Wastes and Plant Chemistry Control

7–4.1 Waste disposal. The radioactive waste disposal system, discussed in detail in Chapter 10, utilizes storage and decay, primarily for gaseous activity, and demineralization for soluble activity. In developing this system, two studies were required to assure that (1) gaseous activities could be collected and stored, and (2) the soluble activities could be effectively removed by demineralization.

A study indicated that existing design correlations were applicable for the stripping of dilute gaseous activities from water. First studied was soluble activity removal by demineralization using carrier-free isotopes. In a specific study of the waste disposal cycle, water from an in-pile test of a reference fuel element was passed through an MB-1 ion exchanger after periods of decay corresponding to the waste disposal cycle. Required decontamination factors were obtained.

7–4.2 Leakage of reactor coolant. Leakage to the plant container and to the boilers is the major problem arising from the presence of fission products and other activity in the coolant. Calculations were made of the concentrations of various isotopes in the plant container air, in the boiler water, and in the effluent to the atmosphere for various numbers of defective fuel elements and assumed leak rates. The results (used in the hazards analysis discussed in Chapter 11) demonstrated that for plausible leak rates (for normal operation), radioactivity thus released was not a limiting factor in waste disposal or operation.

The use of lithium hydroxide in the PWR coolant leads to the formation of tritium. The only possible problem associated with the tritium would occur in the main steam system as a result of leaks in the boilers. An analysis was made of possible tritium activities in the main steam system. It was found that these would be of the order of 0.2 to 0.3 of tolerances given in National Bureau of Standards Handbook 52.

7–4.3 Analytical methods. The PWR coolant sampling system is described in Chapter 8. It permits safe and convenient monitoring of the coolant for operational control, and for the development of information for future pressurized water plants. In view of possible operation with a large number of defective fuel elements, sampling and analytical facilities that minimized personnel radiation exposure were provided. To this end, maximum utilization was made of instrumental methods of analysis incorporated into shield sampling trains. A prototype analytical train was

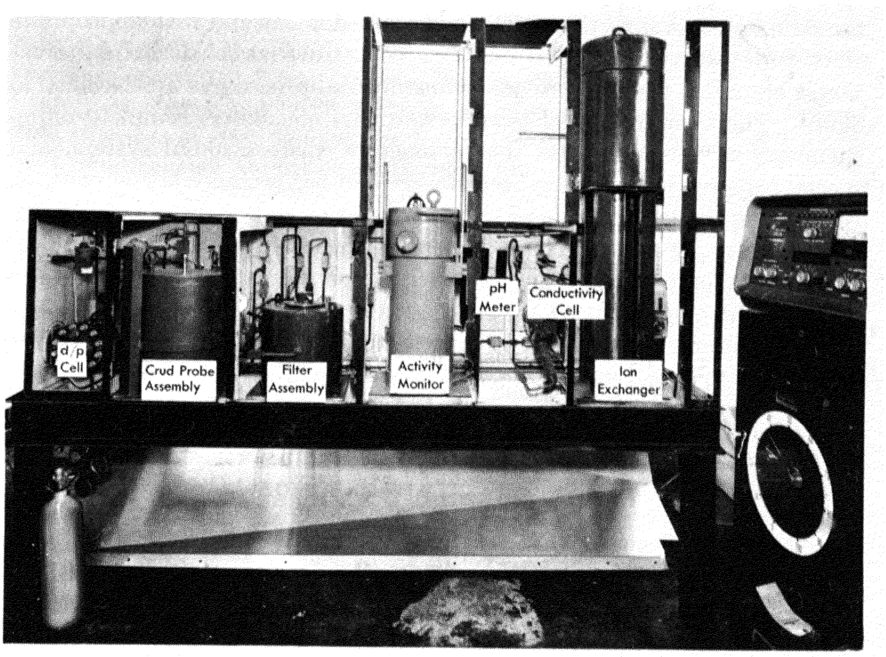

FIG. 7-4. Prototype analytical train.

tested both in the laboratory and with an operating reactor. The type of construction is shown in Fig. 7-4, which is a view of the prototype train showing the individual components.

The Shippingport plant has three analytical trains. One measures the properties of the coolant before it enters the purification system. A second measures the properties of the coolant after it has passed through the purification system. The third is a spare that can be connected to either the inlet or outlet of the purification system.

In addition to providing samples for radiochemical and other analysis, the sampling train measures the following: (1) pH, (2) conductivity, (3) dissolved hydrogen, (4) dissolved oxygen, (5) crud concentration and activity, (6) soluble activity, and (7) accumulated water solubles.

Since hydrogen and oxygen determinations normally account for a large fraction of the personnel radiation exposure received during analysis, special attention was devoted to developing instrumental methods for these analyses. Instruments so developed were incorporated in the sampling train. These are discussed in Chapter 12.

7-4.4 Chemistry test program. The chemical performance of the plant must be checked during operation. To this end the plant is equipped with

the sampling and analytical facilities described above and with appropriate laboratories (discussed in Chapter 16). To assure that the desired information is obtained, testing methods and analytical procedures are specified in detail. These cover normal plant operational chemistry, complete radiochemistry of the plant and the radioactive waste disposal system, and special tests related to coolant technology and performance of components. Included in the latter are tests on recombination of hydrogen and oxygen, effective flux for tritium formation, performance of the coolant purification system ion exchangers, synthesis of ammonia, and buildup of activity on the plant piping.

The facilities provided will be used in the investigation of both the Shippingport Core I and subsequent cores.

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CHAPTER 8

REACTOR PLANT AUXILIARY SYSTEMS

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REACTOR PLANT AUXILIARY SYSTEMS*

Associated with a pressurized water reactor plant are a number of auxiliary fluid systems that are necessary to insure proper and safe operation of the plant. These auxiliary systems can be classified into three functional groups: reactor plant services, reactor plant protection, and reactor plant information.

Under the first category, auxiliary systems are provided to charge high purity water into the reactor plant for initial filling and to maintain pressurizer level, to discharge and subsequently dispose of reactor plant effluents, to purify the reactor coolant, to supply cooling water to reactor plant components, to inject chemicals into the reactor plant in order to maintain the proper water chemistry of the reactor coolant, to operate remotely certain reactor plant valves of major importance, to provide water shielding for core and subassembly transfer and storage, and to remove reactor decay heat following a plant shutdown.

For reactor plant protection, systems are provided for removing reactor decay heat in the event all AC power is lost to the station, for protecting the plant from overpressure surges, and for supplying cooling water to cover the core in the unlikely event of a reactor coolant system pipe rupture. For reactor plant information purposes, systems are provided for sampling the reactor coolant to determine that proper water chemistry is being maintained and to detect and locate a failed fuel element.

8–1. Reactor Plant Services

8–1.1 Coolant charging system. *Function.* The coolant charging system (Fig. 8–1):

(1) Stores reactor coolant grade water at substantially atmospheric pressure.

(2) Furnishes reactor coolant grade water at pressures up to 120 psig for:

(a) Initial filling of the reactor coolant system, pressurizer, pressure relief system, coolant purification system, reactor vessel, and other reactor plant auxiliary systems.

(b) Removing spent resin from, and charging fresh resin to, the purification demineralizer.

(c) Flushing reactor coolant and coolant purification loops.

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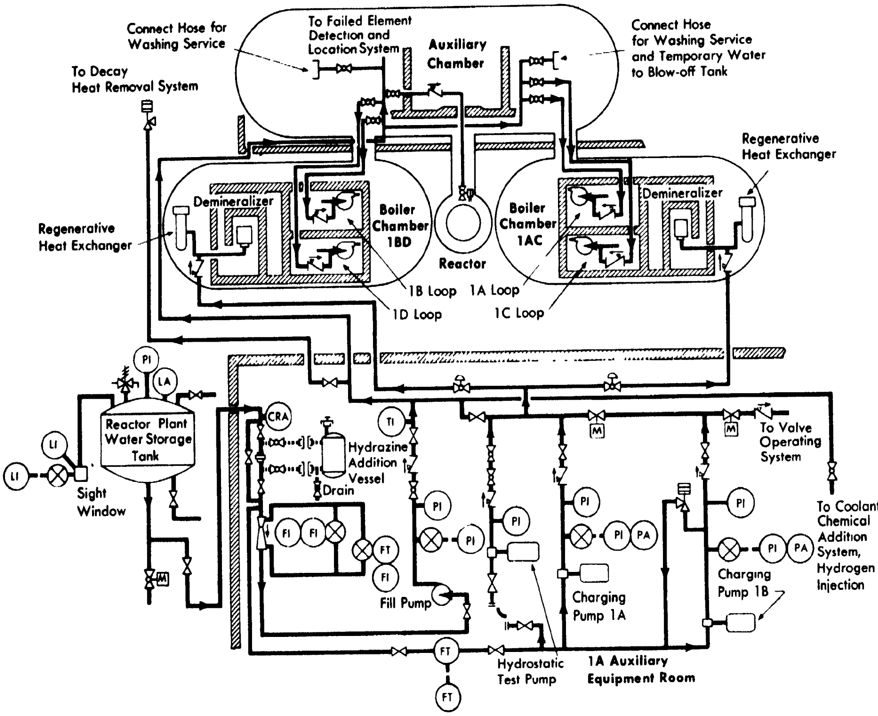
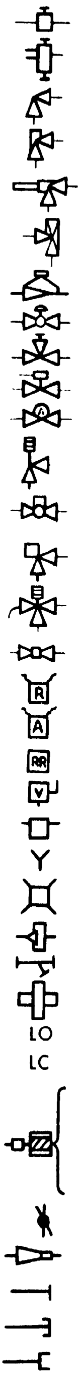


FIG. 8-1. Coolant charging system flow diagram. A portion of the coolant chemical addition system, although not discussed until later, is shown by dotted lines.

The following standard symbols have been used in all the figures in this chapter.

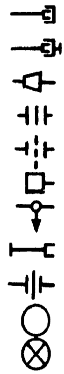
- Gate Valve
- Gate Valve—Capped
- Gate Valve—Motor Operated
- Gate Valve—Hydraulic Pilot Operated
- Butterfly Valve—Hydraulic Pilot Operated
- Main Steam Throttle Valve
- Diaphragm Valve—Saunders Patent
- Globe Valve
- Glove Valve—Capped
- Globe Valve—Motor Operated
- Globe Valve—Hydraulic Pilot Operated
- Reverse Seated Globe Valve—Hydraulic Open, Spring Closing
- Needle Valve—Capped
- Needle Valve
- Solenoid Valve with Hand Control
- Globe Valve—Extended Bonnet
- Globe Valve—"Y" Pattern Body
- Globe Valve—Bellows Seal
- Check Valve—Lift
- Check Valve—Swing
- Check Valve—Excess Flow
- Stop Check Valve
- 3-Way Valve—Solenoid Operated
- 3-Way Valve—Motor Operated



3-Way Instrument Valve
3-Way, 2-Stem Instrument Valve
Angle Valve
Angle Valve—Capped Bonnet
Angle Valve—Extended Capped Bonnet
Angle Globe Valve—Needlepoint Capped
Pressure Reducing Valve
Control Valve—Diaphragm Operated
Flow Control Valve—Hand Operated
Temperature Control Valve
Selector Valve—Air Operated
Angle Relief Valve
Pilot Relief Valve
Angle Relief Valve Pilot Operated
Pressure and Vacuum Relief Valve
Plug Valve
Radioactivity Detector
Radioactive Air Particle Detector
Pneumatic Reversing Relay
Automatic Vent Valve
Sediment Trap
Drain Connection
Floor Drain
Air Filter
Angle-Type Strainer
Basket-Type Strainer

LO	Valve Locked Open
LC	Valve Locked Closed
D	Damper
FD	Face Damper
FAD	Fresh Air Damper
RD	Recirculating Damper
BPD	Bypass Damper

Hand Damper
Venturi
Connection for Temporary Pipe
Pipe—Capped
Hose Connection



Hose Connection Capped
Hand Clamp Connection
Reducer
Orifice Flange Assembly
Equalizing or Restricting Orifice
Air Operated Motor
Flexible Hose Connection
Valve Positioning Tool
Union
Mounted Instrument
Instrument Transmitter

C	Conductivity Cell
CIA	Conductivity Indicator Alarm
CRA	Conductivity Recorder Alarm
CR	Conductivity Recorder
d/p	Differential Pressure Cell (Local Reading)
FI	Flow Indicator
FR	Flow Recorder
FT	Displacement Meter
HC	Hand Control
HR	Hydrogen Recorder
LA	Level Alarm
LC	Level Controller
LG	Level Gauge
LI	Level Indicator
LIA	Level Indicator Alarm
LIC	Level Indicator Controller
LIR	Level Indicator Recorder
LS	Limit Switch
PA	Pressure Alarm
PC	Pressure Controller
PCV	Pressure Control Valve
PDI	Pressure Differential Indicator
pH	pH Cell
pHR	pH Recorder
PI	Pressure Indicator
PIA	Pressure Indicator Alarm
PIC	Pressure Indicator Controller
PIR	Pressure Indicator Recorder
PR	Pneumatic Pressure Regulating Switch
PS	Pressure Switch
psi	Pressure, Pounds Per Square Inch
RC	Reference Chamber
RR	Rate Recorder
TA	Temperature Alarm
TC	Temperature Controller
TI	Temperature Indicator
TR	Temperature Recorder
TRC	Temperature Recorder Controller
RA	Rate Alarm
OR	Oxygen Recorder
TE	Test Element
SC	Sampling Connection

(d) Initial filling and makeup to the blowoff tank, and flushing blowoff and flash tanks in the coolant discharge and vent system.

(e) Initial filling and makeup to the reactor plant component cooling water system, neutron shield tank, and canal water system.

(f) Filling the demineralized water service tank, which serves the laboratories and the clean room.

(g) Flushing sample train instruments and piping.

(h) Flushing the failed element detection and location system delay coils.

(3) Charges reactor coolant grade water into all reactor coolant and purification loops for:

(a) Makeup against system pressure to the reactor plant through each purification loop to maintain the desired pressurizer water level.

(b) Use by the valve operating system and the coolant chemical addition system (at pressure up to 3000 psig).

(c) Pressurization of each reactor coolant and purification loop when isolated.

(d) Pressurization of all other reactor plant systems and the reactor vessel.

(4) Provides reactor coolant grade water at 3750 psig for hydrostatic testing of the reactor plant systems and at 4500 psig for testing of the reactor vessel.

(5) Provides an emergency water supply for the safety injection system.

System flow path. The pumps in the coolant charging system take suction from a common header, gravity fed by the reactor plant water storage tank. This header contains a conductivity probe, a strainer, a flowmeter complete with a volume integrator, a charging water displacement meter, a bypass around the strainer, and a bypass around the charging water displacement meter. Connections in the piping between the reactor plant water storage tank and the pumps provide for tank drainage and for hydrazine addition by the coolant chemical addition system.

The fill pump normally discharges into a "fill" header, and the charging and hydrostatic test pumps normally discharge into a "charging" header. The headers, however, have a valved interconnection to permit the use of either header by the pumps. The fill header is designed for charging pump pressure from the fill pump discharge line through a check valve. A remotely operated valve is located in the charging header, between the charging pump discharge lines, permitting either charging pump to provide reactor plant makeup or to serve the valve operating system. This arrangement provides flexibility in plant operation. In addition, this valve, along with an unloading valve in the line cross-connecting the charging header and the pump suction header, will be used to unload the charging pumps should they ever need to be started by emergency diesel generator power.

Each line from the charging header to a purification loop has a remotely operated valve to direct the flow to the desired loop. There are two such lines, one for each loop.

The high pressure portion of the fill pump discharge line contains a steam mixing nozzle with a temporary steam connection and a temperature indicator. The steam mixing nozzle permits preheating of the fill water used for filling and flushing loops. From the fill header, there is a line to each reactor coolant loop, the pressurizer, the coolant chemical addition system, the coolant purification system test loops, the decay heat removal system, and the demineralized water auxiliary supply stations (one located in each purification and auxiliary chamber valve operating cubicle). These supply stations are each furnished with a valve and a temporary connection and are used for auxiliary services requiring reactor coolant grade water, such as flash tank and blowoff tank flushing, addition of water to the ballast water in the blowoff tank, addition of water to the resin when charging resin into a demineralizer, back-flushing resin in a demineralizer, and external washing of equipment. The low pressure portion of the fill pump discharge line contains a globe valve for throttling the fill water flow rate to the fill header. A low pressure connection in the low pressure portion of the discharge line supplies fill water through a check valve and gate valve to the component cooling water system, the canal water system, and the miscellaneous service lines.

Design requirements. (a) Reactor plant water storage tank. Reactor coolant grade water for the system is contained in an elevated water storage tank. The normal makeup water to the tank is demineralized main condenser condensate supplied at a maximum rate of 15 gpm and a maximum temperature of 120°F. Deaerator water at maximums of 15 gpm and approximately 200°F can be used if main condenser condensate is not available. The steam plant can furnish 6000 gal daily to the tank from either of these two sources after converting river water into high purity boiler feedwater.

The water delivered to the reactor coolant systems meets the following requirements:

(1) Hydrogen ion concentration (pH)
   The pH value of the water must be maintained between 6.5 and 8.3.

(2) Oxygen concentration
   The concentration of oxygen in the water must not exceed 0.009 ppm.

(3) Chloride ion concentration
   The concentration of chloride ions in the water must not exceed 0.3 ppm.

(4) Conductivity

The electrical conductivity of the water is maintained between 0.5 and 1.5 μmhos/cm (corrected to 25°C water temperature).

(5) Nitrogen concentration

The concentration of dissolved nitrogen in the water must not exceed 0.2 ppm.

(6) Concentration of insolubles

The total concentration of insolubles in the water must not exceed 0.2 ppm.

(7) Temperature

The temperature of the water supplied to the coolant charging system must be between 80°F and 120°F.

An overflow from the reactor plant water storage tank leads into a water storage tank in the steam plant which acts as the condensate storage and surge tank. Therefore, a small amount of water circulates through the reactor plant water storage tank and minimizes buildup of impurities in the contained water. Since oxygen in the reactor coolant is undesirable, absorption is minimized by use of a 0.5 psig steam blanket over the water surface of the reactor plant water storage tank. The steam blanket is not essential to reactor plant operation.

The 50,000 gal reactor plant water storage tank has a volume about twice the total volume of the reactor vessel, the reactor coolant system, the pressurizer and pressure relief system, and the coolant purification system. This volume is approximately three times as much as design criteria required for normal conditions.

Excluding the 25,000 gal required for initial fill and 1600 gal required to refill after safety valve discharge, the sizing criteria for the tank are based on the following:

Item	Water consumption, gal/day
Compensation for thermal contraction	4500
Resin discharge	1500
Resin charging	750
Flushing (2 loops)	4000
Refill after flushing (2 loops)	4000
Valve operating system	100
Sampling system (including sample line flush)	130
Miscellaneous (leakage, miscellaneous flushing and fill)	1500
Total	16480

The estimated average charging required by the system in one day is as follows:

Item	Water consumption, gal/day
Sampling system Sample line flushing	130
Valve operating system losses	15
Leakage	280
Total daily requirement	425

(b) Fill pump. The 200 gpm fill pump is used for initial filling of the reactor coolant system and flushing out the main coolant loops; it has adequate discharge pressure (100 psig) for flushing spent purification demineralizer resin to the radioactive waste disposal system. The shut-off head for this pump is approximately 120 psig.

(c) Charging pumps. The combined capacity of the charging pumps, 50 gpm displacement at full speed, is adequate to compensate for the decreased volume caused, during reactor plant shutdown, by the thermal contraction of the water in the reactor, the reactor coolant system, the pressurizer and pressure relief system, and the coolant purification system. One pump is adequate to supply water intermittently to the flask in the valve operating system, and the other to keep the reactor plant charged to its normal content. The discharge pressure of the pumps was determined by the pressure required for the valve operating system, 3000 psig.

(d) Hydrostatic test pump. The hydrostatic test pump has a maximum discharge pressure of 4500 psig, which is equal to 1.5 times the design pressure of the valve operating system. This pump is used to test plant piping at a pressure of 3750 psig (1.5 times the design pressure of the reactor coolant loops and auxiliary systems). For this testing a relief valve, set at 3800 psig, is provided in the discharge piping. This relief valve is gagged when testing is being done at 4500 psig, and another relief valve, set at 4550 psig, is provided to protect against overpressure. The capacity of the hydrostatic test pump is 2 gpm displacement at full speed.

8-1.2 Coolant discharge and vent system. *Function.* The coolant discharge and vent system (Fig. 8-2):

(1) Accepts radioactive fluids vented and drained from the reactor plant, including:

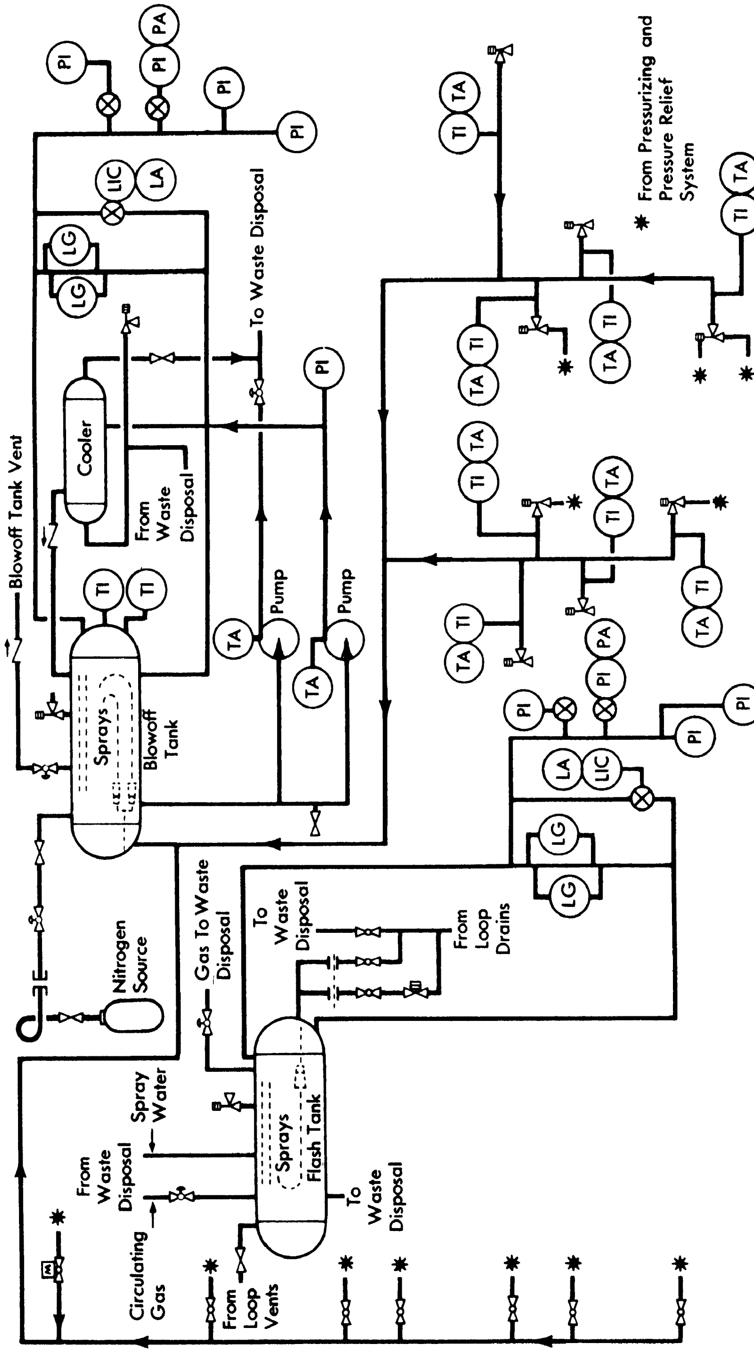


FIG. 8-2. Coolant discharge and vent system flow diagram.

(a) Vent gas from each reactor coolant pump, from the reactor coolant side of each steam generator, from each purification demineralizer, and from the pressurizer.

(b) Drain water from each reactor coolant loop, from each purification loop, from each purification test loop, and from the pressurizer.

(2) Depressurizes and cools accepted fluids and discharges them to the surge and decay tanks in the radioactive waste disposal system.

(3) Accepts and contains relief valve discharge from the reactor vessel, from the pressurizer vessel, from each reactor coolant loop, and from each purification loop.

(4) Depressurizes and cools the influent from valves, and discharges it to the vent gas header in the radioactive waste disposal system.

(5) Accepts hydrogen gas into the blowoff and flash tanks and discharges it to the vent gas header in the radioactive waste disposal system.

(6) Discharges resin from each purification demineralizer to the resin storage tanks in the radioactive waste disposal system.

System flow path. The loop drain piping from each reactor coolant loop contains a remote hydraulically operated isolation valve connected to the low point of the reactor coolant loop. Downstream from each of these valves is a manual stop-check valve. The drainage piping from each purification loop contains an accessible, capped, manual isolation valve connected to the demineralizer outlet piping and a stop-check valve downstream of the isolation valve. (The stop-check valves are normally open, and are closed only for maintenance.) Downstream of the respective stop-check valves, each of the above drain lines joins into a common header. Also connecting into this common header are drain lines from the pressurizer and the purification test loops, each containing an accessible extended cap isolation valve.

Downstream from all the connecting drain lines, the drain header branches into three lines. Drainage may flow through any of the three branches, depending upon the temperature of the drainage. For operating temperatures (508 to 542°F), the drainage is passed through the branch containing a remote hydraulically operated valve (a back-up valve for the hydraulically operated loop drain valve which is normally closed, but opened remotely when this branch is used), an isolation globe valve (normally open; closed only for maintenance), and an orifice. This line terminates at the drain inlet connection to the flash tank. For loop cooldown temperatures (approximately 250°F), the drainage is passed through the branch containing an accessible globe valve (normally closed; opened manually when this branch is to be used) and an orifice. This line terminates, in common with the operating temperature line, at the drain inlet connection to the flash tank. For drainage temperatures of 150°F and lower, the third line pipes drainage directly to the radioactive waste disposal

system without further cooling in the flash tank. This line contains an accessible globe valve, normally closed but manually opened when this branch is to be used. No orifices or other valves are contained in this branch drain line.

The orifice in the 500 to 542°F drain line limits the drainage flow rate to 40 gpm of coolant at 525°F maximum temperature and 2000 psia upstream pressure. The 40 gpm was calculated to be the maximum rate of thermal expansion of the reactor coolant during plant warmup. The orifice located in the 250°F drain line limits the flow rate to 50 gpm of coolant at 250°F maximum temperature and 50 psig upstream pressure. The controlling factor in this case was the maximum hydrogen gas release rate that can be handled by the waste disposal recirculation vent gas. Rates above 50 gpm result in buildup of hydrogen in the flash tank and initiation of the hydrogen concentration alarm.

The primary sides of the steam generators, the reactor coolant pumps, the purification demineralizers, and the pressurizer are vented through lines to the flash tank, which vents to the radioactive waste disposal system.

To prevent the formation of explosive hydrogen-oxygen pockets in the flash tank, approximately 15 standard cubic feet per minute of this vent gas is circulated through the flash tank by the radioactive waste disposal system. To permit isolation of the flash tank from the waste disposal system, the inlet vent and outlet vent lines each contain accessible, noncapped, manually operated gate valves. An air operated diaphragm valve is provided in each inlet and outlet line of the flash tank to protect the radioactive waste disposal system from overpressure. The flash tank is provided with a pressure controller that closes these waste disposal diaphragm valves at 3 psig in the outlet line from the flash tank.

The vent line from the blowoff tank contains a remotely operated diaphragm valve and a check valve. Downstream of the check valve, the vent line, which contains no branches, connects directly into the radioactive waste disposal system.

The blowoff tank is normally under a nitrogen blanket at approximately 2 psig to assist in keeping the tank vapor space oxygen free. The nitrogen originates at a readily accessible location outside of the plant container to permit convenient renewal of the nitrogen gas blanket from stored nitrogen gas bottles.

The discharge lines from each relief valve connected to the reactor and pressurizer vessels join into a common relief valve header that terminates at the relief valve inlet on the blowoff tank. The discharge lines from each relief valve connected to the reactor loops and purification loops join into another common relief valve header which terminates at the relief valve inlet on the blowoff tank. The two relief valve headers

join before connecting to the single relief valve inlet connection on the blowoff tank.

The spray water line from waste disposal contains a remote air operated diaphragm valve for on-off supply of spray water to the flash tank.

The cooling water inlet line from waste disposal contains an accessible, noncapped, isolation gate valve and terminates at the cooling water tube side inlet connection. This line also contains a relief valve between the spray water cooler and the isolation valve to protect from overpressure due to thermal expansion when isolation valves are closed. The cooling water outlet line originates at the tube side outlet connection and contains an orifice to limit the maximum cooling water flow rate. Downstream of the orifice, the line contains an accessible, noncapped, isolation gate valve before connecting into the radioactive waste disposal system.

The resin discharge lines from each purification demineralizer contain an accessible, noncapped, manually operated globe valve which normally isolates the demineralizers from the radioactive waste disposal system. Downstream of each of these valves is an accessible, noncapped, manually operated plug valve. Located between the isolation valve and the backup valve is a back-flush line containing a strainer, a gate valve, and a connection for a water hose (water supplied by the coolant charging system). This line is used for back-flushing the isolation valve disc and seat after the discharge of resin to waste disposal.

All piping and component connections, including valves in contact with reactor coolant, are butt welded or socket welded. An exception is permitted downstream of the first valve isolating the coolant discharge and vent system from a system containing reactor coolant. In such cases, other type connections are desirable for ease of maintenance, but these are to have essentially zero leakage to the reactor plant container.

Design requirements. All drain and vent lines connected directly to the high pressure reactor system contain two valves, in series, to minimize leakage of reactor coolant across the closed valve seats. Drain and vent lines from the high pressure reactor coolant systems that discharge to the reactor plant container each contain a stop valve. The end of each valve outlet is capped with a blind flange to minimize leakage of reactor coolant or gas to the reactor plant container.

To avoid possible contamination of the steam system by reactor coolant leakage across valves, steam system drains and vents are not connected in any manner to the coolant discharge and vent system piping.

Remote and automatic operations have been minimized. All remotely operated air diaphragm valves have manual overrides. The three way motor operated control valves for the hydraulically operated drain valves can be manually operated. Hydraulically operated valves are used because they can be hermetically sealed, assuring zero leakage to the reactor plant

container, and can be opened or closed even if all electrical power is lost.

The coolant discharge and vent system can be operated remotely from outside the reactor plant container when the system is used for pressurizer level adjustment. Manual adjustments for other system operations, requiring access to valves and hose stations inside the reactor plant container, are minimized by use of remote controlled valves.

(a) Spray water cooler. The spray water cooler has sufficient capacity to cool in $1\frac{1}{2}$ hr the contents of the blowoff tank, following the maximum relief valve discharge, to the approximate normal ballast water temperature of 125°F. The additional heat exchanger surface necessary to cool the ballast water to the normal temperature of 120°F in this time was not justified by the advantage to be gained.

(b) Flash tank discharge pump. The flash tank discharge pump (and spare), with a capacity of 300 gpm at 35 psi discharge head, is adequate to maintain the flash tank ballast water level within the operating high and low ballast water level range. The pump will not cycle on and off when an isolated loop is drained into the flash tank at the rate of 50 gpm with a flash tank spray rate of 250 gpm.

(c) Flash tank. The flash tank has enough ballast water for depressurizing and cooling 40 gpm of 525°F reactor coolant to adjust the pressurizer level from its high level to its low level alarm setting (28 ft^3) without spraying water into the flash tank. To assure adequate mixing of the drain water with the ballast water for this condition, a mixing nozzle inside the tank is connected to the drain inlet connection. It is located under the ballast water (beneath the low level alarm setting) to insure efficient mixing. The flash tank also has sufficient capacity, with the spray water capacity of 20,000 gals, to permit continuous drainage of 525°F reactor coolant at 1985 psig at a maximum rate of 40 gpm, or 250°F reactor coolant at 50 psig at a maximum rate of 50 gpm.

Hot reactor coolant vent gases must be vented to the vapor space to be depressurized and cooled. The geometrical shape and size of the vapor space are suitable (a) for efficient spray water dispersion through the spray nozzles attached to the tank spray water header, and (b) for ingress and egress of low hydrogen concentration waste disposal vent header gas which prevents the formation of an explosive hydrogen-oxygen mixture.

(d) Blowoff tank. The blowoff tank has sufficient capacity, together with the spray water cooler and blowoff tank discharge pump, to depressurize, cool, and discharge to waste disposal the maximum relief valve effluent (192 ft^3 total accumulation at a maximum rate of 3.2 ft^3/sec) that may be discharged to the blowoff tank by the reactor plant. The blowoff tank can be restored to normal operating conditions within two hours after receiving the maximum relief valve discharge.

Two mixing nozzles quickly depressurize, cool, and adequately mix the relief valve influent with the blowoff tank ballast water. These nozzles join inside the tank and connect to a common relief valve influent connection. They are under the ballast water (beneath the low level alarm setting) to ensure efficient mixing.

The geometrical shape and size of the vapor space are suitable for efficient spray water dispersion through the spray nozzles attached to the tank spray water header, and for the acceptance of steam that may be flashed and gases that are stripped from the relief valve influent. The blowoff tank must be kept oxygen-free to prevent the formation of an explosive hydrogen-oxygen mixture. This is accomplished by a nitrogen gas blanket under a slight positive pressure.

(e) Blowoff tank discharge pump. The blowoff tank discharge pump (and spare), with a capacity of 400 gpm at 50 psi discharge head, is a duplicate of the flash tank pump except for the additional capacity. This pump is adequate for maintaining the blowoff tank ballast water level within the operating high and low ballast water level range when the pump discharge valve is open to permit water discharge to waste disposal. If the pump discharge valve is closed, the pump has adequate capacity to discharge 100 gpm at 65 psi discharge head through the blowoff tank spray water cooler and the blowoff tank spray header for recirculation of ballast water.

(f) Spray water. Spray water from the radioactive waste disposal system is partially processed water which has been cooled to 100°F by that system for use in the flash tank as spray water and as cooling water for the blowoff tank spray water cooler. The use of this water is satisfactory for this system and reduces the capacity requirements of the radioactive waste disposal system.

8–1.3 Coolant purification system. *Function.* The coolant purification system (Fig. 8–3):

(1) Removes dissolved and suspended solid impurities from the reactor coolant.

(2) Maintains the reactor coolant at its nominal pH value.

(3) Provides a means for monitoring the build-up of residual radioactivity on the inside surfaces of the reactor coolant systems.

(4) Provides a means for determining the dissolved hydrogen gas concentration in the reactor coolant.

General description. The coolant purification system has two identical loops, each capable of purifying the entire reactor coolant system. Each loop has two heat exchangers, one regenerative and one nonregenerative, and a demineralizer. One purification loop is connected to two reactor coolant loops by a valving arrangement which allows purification influent

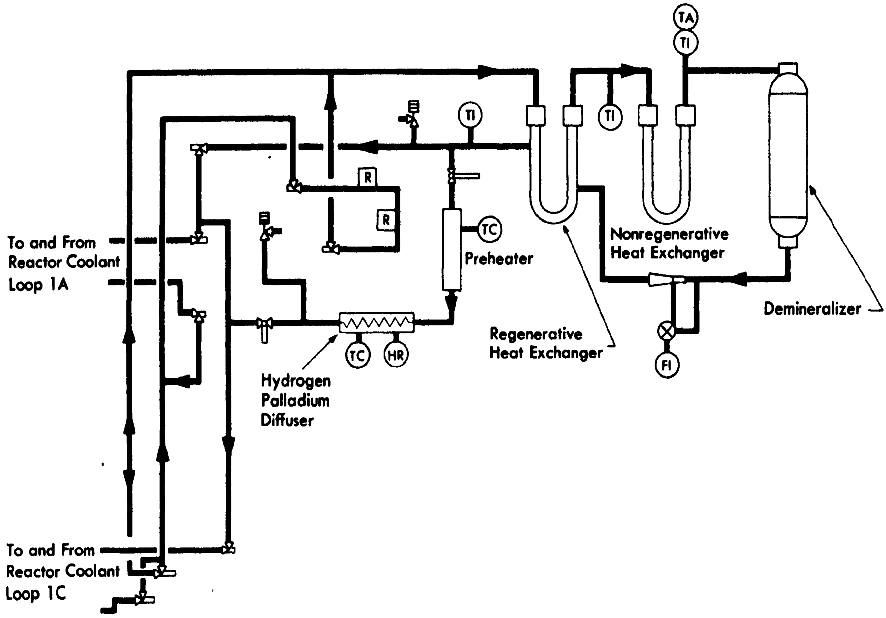


FIG. 8-3. Coolant purification system flow diagram.

to be drawn from either coolant loop and effluent to be discharged to either coolant loop and, in addition, satisfies the double isolation valve requirement of the high pressure reactor coolant system.

In normal operation, the coolant purification system bleeds a portion of the reactor coolant at a controlled rate to each of the two purification loops from the discharge sides of the reactor coolant pumps feeding the purification loops. In each loop, the coolant passes through a set of manual, capped, isolation valves to the contamination monitoring station, consisting of a "hairpin loop" called the purification test loop. Flow is thence to the tube side of the regenerative heat exchanger and through the tube side of the nonregenerative heat exchanger. The coolant then flows through the demineralizer, back through the regenerative heat exchangers on the shell side, through a set of manual, capped, isolation valves, and returns to the suction side of the reactor coolant pump to which the loop is connected. A small stream of coolant in each purification loop is bypassed through a dissolved hydrogen analyzer, located downstream of the shell side of the regenerative heat exchanger.

The demineralizers remove dissolved impurities and filter suspended particulate matter. The regenerative heat exchangers partially cool the influent reactor coolant and transfer the heat to the effluent coolant stream, thereby minimizing thermal losses and preventing excessive thermal stress

in the return lines. The nonregenerative heat exchangers cool the reactor coolant to a temperature compatible with the thermal stability of the demineralizer resin.

Each of the purification loops is in a limited access area within the boiler chambers of the reactor plant container. These areas are identified as the 1AC and 1BD purification cubicles. Most system piping and valving and the two heat exchangers for each loop are beneath the floor of the purification cubicle. The demineralizer is in a separately shielded concrete cubicle. The purification test loop and the dissolved hydrogen analyzer loop are both in a steel walled cubicle within each purification cubicle.

Design requirements. (a) Impurity removal. The coolant purification system can remove impurities from the reactor coolant without decreasing the effectiveness of any chemical additives, such as corrosion inhibitors, which may be present in the system. Impurities from two sources are removed: those due to corrosion of piping, components, and core cladding; and those due to the release of fuel material after a failure of the cladding. The following are purification design data and assumptions:

(1) A maximum corrosion rate of loop material and cladding of 10 mg/dm^2/mo.

(2) A maximum fuel element failure rate of 1000 per core life.

(3) A surface area of 35,000 ft^2 exposed to coolant.

Dissolved and suspended impurities are removed from the coolant to maintain the following requirements:

(1) The activity of the volatile and nonvolatile radioactive material in the coolant must not exceed 1×10^7 dis/sec·cm^3. This is consistent with the secondary shield design, which limits the exterior maximum dose rate to 5.0 mr/hr in areas not normally occupied but accessible and 2 mr/hr when the surface bounds a normally occupied area. So that very high loop decontamination factors after shutdown would not be necessary, this value is kept as low as possible consistent with a purification system of reasonable size.

(2) To inhibit crud deposition and fouling of heat transfer surfaces in contact with reactor coolant, the amount of suspended impurities in the coolant must be no greater than one ppm. Particulate matter must be removed from the coolant to prevent plugging of coolant flow passages through small clearances in the core and system components.

The soluble product ion exchange process taking place in the lithium hydroxide resin in the demineralizers serves to maintain the reactor coolant at its nominal pH value of 10.0.

A special section of piping is included in the system to provide a means for determining the rate of contamination buildup on reactor plant inside surfaces in both vertical and horizontal planes. This is called the purification test loop.

A portion of the purification flow steam is bypassed just before it re-enters the reactor coolant system. Equipment in this bypass loop obtains data for recording the dissolved hydrogen gas concentration in the reactor coolant.

To prevent damage to the resin bed, reactor coolant entering the coolant purification system at approximately 508°F is cooled to 120°F by a two stage cooling process before it enters the resin bed.

Coolant returning to the reactor coolant system is reheated to the maximum economic temperature attainable with practical regenerative heat exchanger design. This temperature is greater than the minimum designated by equipment thermal shock requirements.

Facilities are provided in the coolant discharge and vent system for discharging coolant purification system spent resin to the 1A and 1B resin storage tanks of the radioactive waste disposal system.

There is adequate shielding for all portions of the system where high levels of radioactivity may concentrate.

(b) Reactor coolant loop warmup. Each purification loop is interconnected and valved between two reactor coolant loops so that the system can be used for warming up at a controlled rate a cold coolant loop from a hot loop.

The core removal cooling system utilizes the nonregenerative heat exchanger and part of the purification loop piping when it is in operation. The coolant charging system connection to each purification loop is used for normal additions of reactor coolant makeup water. The coolant sampling system piping is connected to the purification loops so as to allow sampling of reactor coolant before and after purification.

(c) Chemical addition. The coolant chemical addition system utilizes the coolant sampling system piping, which connects to the purification loops, as a route for hydrogen gas injection into the reactor coolant system.

(d) System design conditions. Each purification loop normally passes 20,000 lb of water per hour and is designed for a maximum water flow of 25,000 lb/hr at 2000 psia. Coolant enters the system at 508°F, is cooled to 120°F prior to entry into the demineralizer, and is reheated to 435°F before returning to the reactor coolant system. All mechanical components in the system are designed for a pressure of 2500 psig or greater.

8–1.4 Coolant chemical addition system. *Function.* The coolant chemical addition system provides facilities for adding to the reactor coolant system (1) oxygen scavengers and corrosion inhibitors, (2) lithium hydroxide solution, and (3) a neutron poison or neutron absorbing substance. (A portion of the system is shown in Fig. 8–1, while the remainder is shown in Fig. 8–11.) Oxygen must be removed from the reactor coolant because it promotes corrosion and, in some cases, may increase the torque requirements of mechanisms. Addition of lithium hyroxide to

the coolant raises its pH, thereby minimizing crud deposition in the core and reducing the corrosion rate and corrosion product release from the stainless and carbon steel portions of the plant. Neutron poison must be added to the reactor coolant if a cold shutdown of the reactor is required and the reactor control rods cannot, for some reason, be inserted.

General description. (a) Oxygen removal. Oxygen is removed from the reactor coolant system in two ways: (1) by introducing pure gaseous hydrogen into the system during startup or normal plant operation; or, (2) if sufficient residual radiation has not built up to catalyze the reaction of hydrogen with oxygen, by injecting an aqueous solution of hydrazine into the system.

A bank of three hydrogen cylinders connects to the hydrogen injection vessel and associated piping and valving (Fig. 8-11). This equipment is in the cylinder storage area of the sample room, which is adjacent to the 1AC boiler chamber. With the exception of the hydrogen cylinders, the equipment is housed in a heated enclosure to protect it from freezing. Hydrogen is admitted to the reactor coolant system via the coolant sampling system and the coolant purification system piping. It can be added directly from the hydrogen cylinders or indirectly through the hydrogen injection vessel, depending on the reactor plant pressure at the time.

A predetermined amount of hydrazine is placed in the small hydrazine addition vessel located in the 1A auxiliary equipment room on the suction side of the coolant charging system pumps (Fig. 8-1). From the vessel the hydrazine solution is pumped through the charging system low or high pressure headers to any of the four reactor coolant loops. The purification system is isolated during this time.

(b) Raising coolant pH. Lithium hydroxide solution is added to the reactor coolant system during initial filling. The solution is sufficiently strong to attain a coolant alkalinity of pH 10. The demineralizers then automatically maintain the pH in the desired range. Subsequent additions of the solution can be made if the coolant pH ever drops to below 9.5. If the coolant pH exceeds 10.5, pure water may be fed into the plant and a corresponding amount of high pH coolant discharged, thereby reducing the over-all coolant pH.

The lithium hydroxide solution is added in the same manner as the hydrazine solution previously noted. A predetermined concentration of a solution of lithium hydroxide monohydrate is placed in the hydrazine addition vessel and pumped into the plant by means of the coolant charging system pumps.

Addition of neutral pH water from the reactor coolant storage facilities to replace coolant discharged during venting, sampling, and other coolant drain procedures tends to lower the coolant pH. This action is balanced by soluble product ion displacement in the purification system demineral-

izers, which adds lithium and hydroxyl ions to the coolant, thereby tending to raise its pH.

(c) Cold chemical shutdown. If one or more reactor control rods should become stuck out of the core and the reactor can be maintained at a controlled criticality but cannot be shut down safely to a cold subcritical condition, chemical shutdown would be necessary. The shutdown can be accomplished by injecting a neutron absorbing boric acid solution into the reactor coolant system.

The boric acid solution is prepared in the resin slurry tank of the radioactive waste disposal system, in the 1A auxiliary equipment room. The tank outlet connects to the charging system suction header, and the boric acid solution is pumped into the reactor coolant system by one of the charging pumps.

After a chemical shutdown, the borated coolant is removed from the reactor coolant systems by a bleed-and-feed process. Final reduction of the boric acid concentration is accomplished by the purification system ion exchangers.

Design requirements. (a) Oxygen removal with hydrogen. Although oxygen is continuously formed by decomposition of water in the reactor, the reverse reaction (induced by a gamma radiation field with an excess of hydrogen) consumes the oxygen as fast as it is formed, resulting in a steady state oxygen concentration well below the 0.14 ppm specified maximum. For this reason, a residual amount of hydrogen is desired in the coolant during operation. Hydrogen can be added during (1) precritical operations before raising coolant temperature to 220°F, (2) startup after initial criticality, and (3) normal reactor plant operation to maintain the hydrogen concentration required to inhibit corrosion. The system can also replenish, during normal plant operation, the hydrogen lost with discharged coolant and by diffusion to the pressurizer steam space.

To inhibit corrosion, hydrogen concentration of 25 to 50 cc (at standard temperature and pressure conditions) per kg of coolant is required for coolant temperatures over 220°F. The components of the hydrogen injection portion of the system require design conditions of 3000 psig at 220°F.

(b) Oxygen removal with hydrazine. Oxygen also appears in the reactor coolant because the coolant dissolves some of the air in the loops during filling.

All precritical oxygen removal operations must be accomplished with hydrazine, because the reaction of oxygen with dissolved hydrogen gas takes place effectively only in a gamma radiation field. Hydrazine can also be added to the coolant to assist the gamma burnout of oxygen so that removal can be rapidly accomplished after a loop filling operation.

Enough hydrazine must be added to the reactor coolant system to reduce

the oxygen content of the coolant to the permissible limit of 0.1 cm^3/liter (0.14 ppm). The components of the hydrazine addition portion of the system require design conditions of 25 psig at 220°F.

(c) Maintenance of high coolant pH. The pH of the reactor coolant system is maintained within the range of 9.5 to 10.5. The nominal pH of the coolant is 10.0.

Coolant pH is measured by means of the coolant sampling system. The analytical trains in the sampling system incorporate pH cells whose outputs are recorded on a multipoint recorder.

The monohydrate form of lithium hydroxide is used. These crystals are deliquescent; in inventory, they are kept sealed in one pound jars or bottles. Approximately four pounds of lithium hydroxide is a one-year supply, assuming a normal plant make-up rate of 1000 ft^3 of coolant per month and a small contingency for loss, spillage, and waste of the chemical.

Since the components used for lithium hydroxide solution addition are also those used for hydrazine addition, system design conditions are 25 psig at 220°F.

(d) Cold chemical shutdown. Poisoning the reactor by adding boric acid to the coolant because one or more control rods are stuck out of the core is justifiable only if all means of regaining control of the malfunctioning rods have failed.

Sufficient boric acid solution is required to produce a cold subcritical shutdown of at least 2% margin of reactivity. Obviously, it is impossible to determine before the fact the number of control rods that may become stuck out of the core. The most severe case of all control rods stuck out of the core has been selected for purposes of system design.

8-1.5 Reactor plant component cooling water system. *Function.* The reactor plant component cooling water system removes heat from various components in the reactor plant in a safe, controllable manner, transferring it to the water of the circulating water system. The cooling water system also supplies the small quantities of corrosion inhibited makeup water required by the neutron shield tank during plant operation.

System flow path. The major components of the system (see Fig. 8-4) are:

(1) Two motor driven recirculating pumps.

(2) A cooler (tubular heat exchanger) to transfer the heat absorbed by the system to the river water.

(3) A number of parallel cooling circuits or loops in which heat is absorbed from various components in the reactor plant systems.

(4) A normally vented expansion tank.

The cooling water flows in a closed circuit through the components being cooled. Flow sequence is in the order listed above, with the recirculating pumps receiving the water from the expansion tank. The parallel cooling

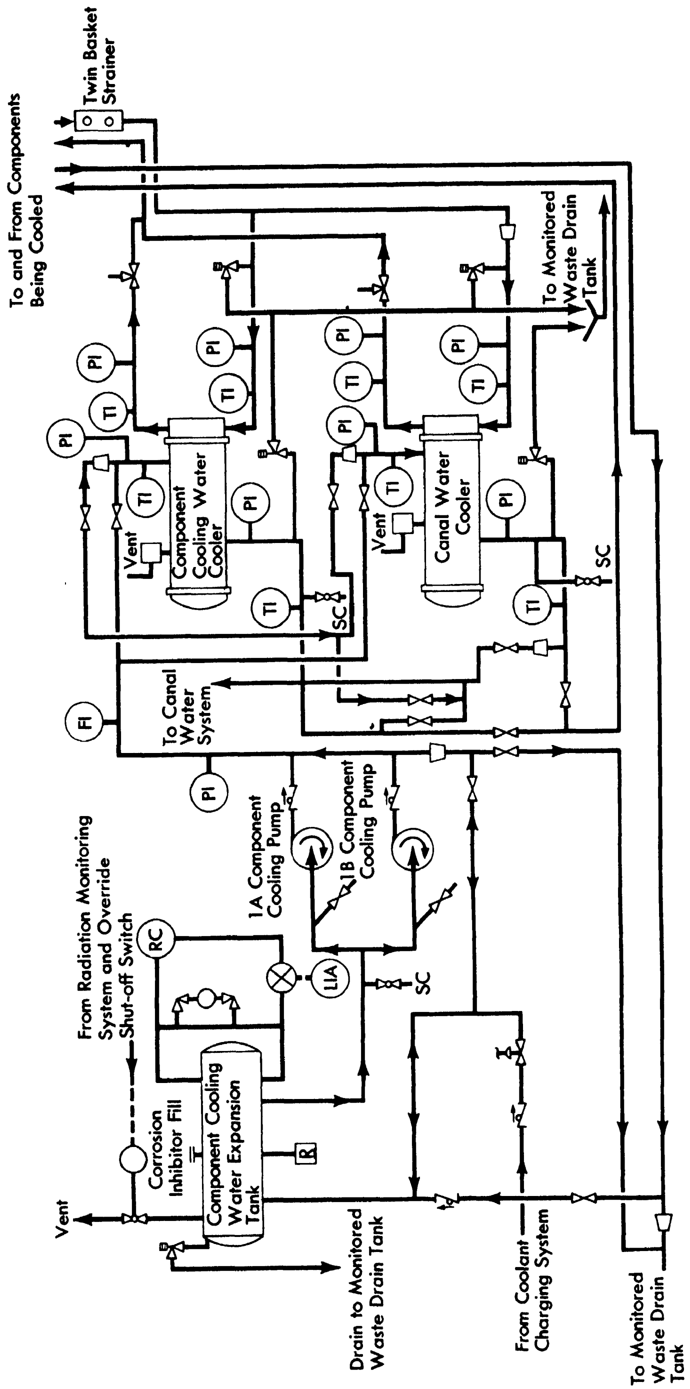


FIG. 8–4. Reactor plant cooling water system flow diagram.

loops are connected to branch headers which, in turn, are connected to two main headers. These are the supply header, running from the cooler to the branch headers, and the return header, returning from the branch headers to the expansion tank.

(a) Components cooled by system. The system removes heat from the following components of other systems:

(1) The four canned pumps of the reactor coolant system.

(2) The two nonregenerative heat exchangers of the coolant purification system.

(3) The four canned pumps of the coolant discharge and vent system.

(4) The high pressure air compressor of the valve operating system.

(5) The two monitors, the canned pump, and the multiport valve of the failed element detection and location system.

(6) The neutron shield tank.

(7) The 32 control rod drive mechanisms.

(b) System fill and makeup. Initial filling and makeup is supplied by the coolant charging system fill pump. The charging system is supplied with demineralized condensate from the main condenser, and with filtered, softened, demineralized, and deaerated river water. Corrosion of the component cooling water system components and piping is minimized by adding potassium chromate, thereby maintaining the chromate concentration in the system water between 500 and 1000 ppm.

Design requirements. (a) Temperature and pressure. The maximum normal operating temperature of the water entering the cooler is 126°F, and the water leaves the cooler at 100°F for a total heat load of 10,127,000 Btu/hr. The design pressure for the system, except for the cooling water jackets for the valve operating system air compressor, is 150 psig. The design of the cooling water side of the air compressor is 70 psig. Excessive water pressure in the system is taken care of by relief valves.

(b) Cooler. The load for the cooler is 10,530,000 Btu/hr for a component cooling water flow of 822 gpm and a river water flow of 1000 gpm with a fouling factor of 0.000256 $Btu^{-1} \cdot hr \cdot ft^2 \cdot {}^\circ F$.

A second cooler in the canal water system is identical to the component cooling water cooler, except that the shell side is rubber lined because there is no corrosion inhibitor in the canal water. If the cooler normally used by this system were unavailable for service, the canal water cooler could be used; the converse is also true. If a spent core is in the canal at the time, the canal water system can cool it for at least 40 hours by recirculating its large volume of water past the core. When the canal water temperature reaches 120°F, the cooler should be returned to canal water service even if the plant must be shut down. It is expected that the unavailable cooler can be repaired for service in this period of time. The capacity of each cooler is considerably greater than that required for the canal water system, so that each can act as a spare for the other.

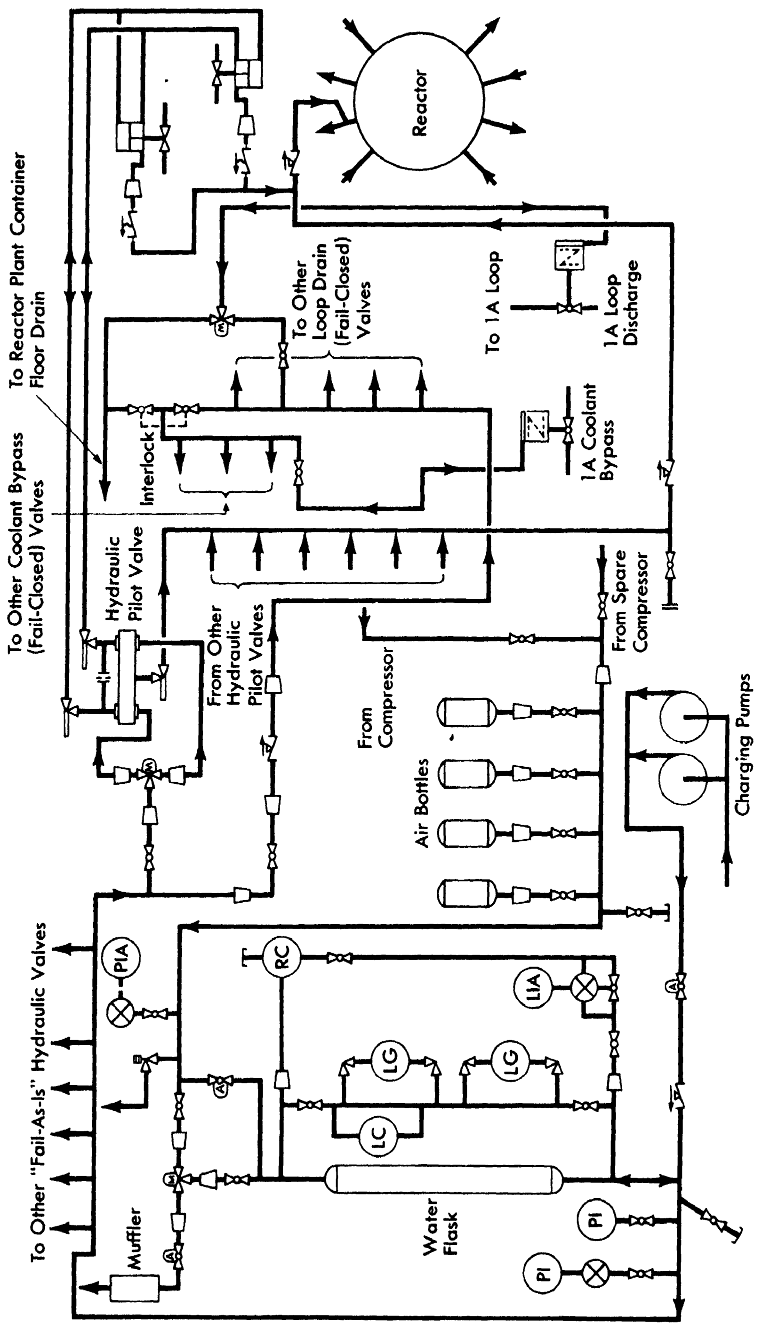


FIG. 8-5. Valve operating system flow diagram.

Each of the cooling water recirculating pumps has the capacity to recirculate 900 gpm at 250 ft total developed head. This head allows for pressure drop through the piping, the flow control valves, the components being cooled, the expansion tank, and the cooler. Strainers protect the recirculating pumps, the cooler, and the cooling passages of the components being cooled. The flow of water in each loop can be measured and regulated to meet the cooling demand. Cooling water temperature indicators in the return lines of each cooling loop and in the supply branch headers to each container can, in conjunction with the flow measuring devices provided, allow determination of the heat load in each loop. The pressure loss in each loop can be determined by installing test indicating pressure gages in valved taps on the supply and return lines.

(c) Leakage. Leakage of component cooling water out of the system, or leakage of external water into the system, can be determined by a continuous decreasing or increasing variation of the water level in the expansion tank. Leakage of radioactive coolant into the system is determined either by level increase or by an alarm from a thimble-enclosed radiation detector in the expansion tank. Coolant leakage into the system can also be located by abnormally high water temperature in the system and by sampling the cooling water returned from each loop through a valved tap on each return line.

(d) Drainage. After determining the degree of radioactivity by sampling the system water, the system can be drained or pumped either to the sewage system or to the radioactive waste disposal system. Drainage within the container is handled by the reactor plant container gravity drain system.

(e) Corrosion inhibitor. Potassium chromate inhibiting solution can be added a batch at a time to the water in the expansion tank. This inhibitor will minimize corrosion of the cooling passages in the components being cooled as well as in all system components and piping.

8-1.6 Valve operating system. *Function.* The valve operating system (Fig. 8-5) operates remotely and with the highest reliability certain reactor plant valves of major operating and safety importance. Direct access to these valves, which are within the reactor plant container, is impossible when the reactor plant is operating because of the high level of radiation. The valves served by this system are component parts of the reactor coolant system, the pressurizer and pressure relief system, the coolant discharge and vent system, and the failed element detection and location system.

General description. (a) Valve actuators. The types of valve actuators operated by this system are (1) fail-as-is and (2) fail-closed on loss of hydraulic motive power. Fail-as-is actuators are of the hydraulic cylinder

and double acting piston type. Fail-closed actuators are of the hydraulic cylinder and single acting piston type to open and are spring-loaded to close. Actuators are integral parts of the valves.

(b) Valves. The system operates the fail-as-is type actuators of the following valves: eight main stop valves at the inlets and outlets of the reactor vessel, the pressurizer surge line stop valve, the pressurizer spray line globe valve, and the failed element detection and location system isolation globe valve. Actuators for the two main stop valves in each reactor coolant loop are under a single control; they act together as a single unit in both opening and closing, although there may be some lag between their strokes. The system also operates fail-closed type actuators of the following valves: four reactor coolant loop bypass valves, four reactor coolant drain valves, and the flash tank inlet valve. All are identical reverse seated globe valves.

(c) Power source. The hydraulic accumulator type system utilizes nonradioactive, high purity water from the coolant charging system, pressurized during operation by high pressure air contained in four air bottles. These bottles constitute the stored air portion of the accumulator; they are kept under pressure by a motor driven air compressor. The pressurized air acts on the water in the water flask to apply hydraulic pressure to the main valve operating supply header. From this header, the high pressure water is directed by suitable valving to the pistons of the various valves served by this system.

The water flask is filled by a pump of the coolant charging system. This is normally the 1A charging pump, although the 1B pump may be used when necessary. The flask is normally vented to atmosphere. To operate a hydraulic valve, the flask vent is closed and high pressure air from the air bottles is admitted through the top of the flask to the water surface, transferring hydraulic pressure to the actuators of various valves. The flask is normally under air pressure only when the hydraulic system must be pressurized to operate a valve or to hold it open, as the type of valve actuator requires. Venting and admitting air to the flask is controlled by a DC motor-operated, three-way air loading valve.

(d) Valve operating methods. Hydraulic fluid displaced by operating the actuating piston of a fail-as-is valve is discharged through a check valve into the reactor outlet line of the 1B loop of the reactor coolant system on the reactor side of the hydraulically operated stop valve. The actuating piston is operated by the differential pressure between the valve operating system and the reactor coolant system. These fail-as-is valves must remain in either the closed or open position for long periods; they are held in position mechanically rather than hydraulically because it is undesirable to continuously charge water into the reactor coolant system.

The actuating piston of a fail-closed valve is operated by the differential

force between the high pressure water acting on the hydraulic piston and the combined opposing force of the valve springs upstream and the hydraulic pressure in the valve body. When a valve of this type is to be closed, the high pressure valve operating water is vented to a floor drain in the auxiliary chamber and thence to the reactor plant container gravity drainage system. The valves may be held open for periods of two hours or more, during which some loss of operating water must be accepted. However, back seats on the actuating pistons keep the leakage negligible.

(e) Valve operational controls. The fail-as-is valves are remotely controlled and actuated through DC motor-operated, three-way selector valves. The motor operators are powered by the reactor plant control batteries. These selector valves operate hydraulic pilot valves to pressurize one side of the operating piston and vent the other side to the reactor coolant system.

Fail-closed valves are controlled and actuated by remotely controlled, DC motor-operated, three-way selector valves without pilot valves and by local manually operated control valves. Both methods admit water to the release water from one side of the valve actuating piston. The former method is used to actuate the reactor coolant loop discharge and the flash tank inlet valves. The latter is employed to actuate the reactor coolant loop bypass valves.

Design requirements. The valve operating system:

(1) Provides remote positive actuation of all vital operating valves under normal and emergency conditions.

(2) Utilizes the most reliable source of electrical energy, the reactor plant control DC storage batteries, for moving the three-way, motor-operated selector valves.

(3) Provides a source of stored energy to move the valve actuators.

(4) Minimizes the manipulations required for valve actuators.

(5) Provides for the isolation of the selector valve and/or pilot valve serving any of the valve actuators in a loop without affecting the ability to operate the other three loops at power.

(6) Utilizes hydraulic fluid compatible with the reactor coolant and the materials and clearances of the equipment used.

(7) Minimizes the oxygen concentration in the hydraulic fluid.

(a) Pressure. The design air pressure of 3000 psig provides for valve operation a reasonable excess over the reactor coolant operating pressure of 2000 psig. The 1000 psi differential pressure allows approximately 450 psi for valve piston differential pressure; 150 psi for line pressure drop; 235 psi for air expansion during operation; and 165 psi for air pressure variation, relief valve tolerance, and high pressure cut-off tolerance.

(b) General water volume. The water volume requirements of the system are based on three sets of conditions, two of which are assumed to be normal and the third constituting an emergency:

(1) Under normal conditions, it may be desirable once a month to isolate one reactor coolant loop; open its bypass valve, its discharge valve, and the flash tank inlet valve; and flush the loop for a period of two hours. The loop is then returned to service.

(2) Under normal conditions, it is necessary at least once a week to operate each valve over a partial stroke and return it to its normal position. This is done to minimize the possibility of corrosion of the valve internals.

(3) Under the emergency condition of a reactor coolant system leak with all four reactor coolant loops in operation, the system must be capable of isolating the pressurizer and each of the main loops. Two of the loops and the pressurizer are then restored to service.

(c) Air capacity. Air storage requirements of the system are based on conditions 1 through 3 above and the following criteria:

(1) Any single, normal, full stroke, hydraulic valve operation, with an initial air pressure of 2835 ($\pm$ 20) psig, must maintain a minimum air pressure of 2600 psig after the stroke is completed.

(2) With an initial air pressure of 2835 ($\pm$ 20) psig, all valve operations required in an emergency must not reduce the air pressure to less than 2000 psig. (It is assumed that reactor coolant system pressure drops to 1000 psig during the emergency, thereby providing 1000 psi differential for emergency operation of the valves.)

8–1.7 Core removal cooling system. *Function.* The core removal cooling system (Fig. 8–6) can remove decay heat from the reactor core during that portion of core or subassembly removal when the core is in the reactor vessel. Although the system is not required for protection of the core during these operations, it is needed to prevent water in the reactor from boiling during refueling operations. Boiling is undesirable for three reasons. First, particulate radioactive material may be dislodged from the reactor vessel surfaces by boiling and then swept into the canal when the core is covered with canal water. Second, boiling results in undesirable working conditions, such as high humidity during the refueling process, and could result in release to the atmosphere of small quantities of fission gases that would normally remain in solution in the reactor coolant. Such release of fission gases would increase the ingestion hazard during these operations. Third, boiling might cause increased corrosion of metal surfaces in the area around the reactor vessel and in the core, masking the condition of the core resulting from operation alone.

The core removal cooling system will assume the decay heat load as soon as coolant loop conditions allow effective heat transfer from the reactor. The system will continue to dissipate the decay heat until the core has been transferred to the fuel handling canal, or in the case of subassembly removal until the reactor coolant pumps have been returned to service.

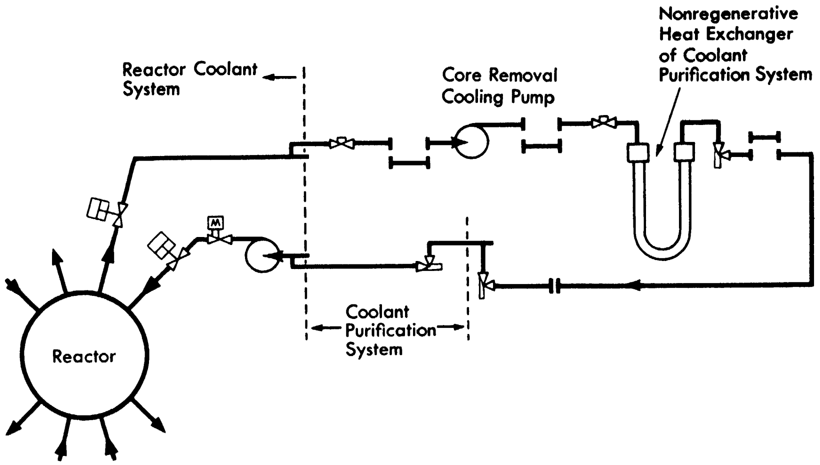


FIG. 8-6. Core removal cooling system flow diagram.

General description. The core removal cooling system utilizes the nonregenerative heat exchangers in the coolant purification system to dissipate the decay heat generated during core or subassembly removal operations. The system driving force is supplied by two centrifugal pumps. Each pump circulates the coolant through one of the nonregenerative heat exchangers, where decay heat is removed by component cooling water. The cooled water then circulates to a reactor inlet nozzle, upward through the core, and through a reactor outlet nozzle to the pump suction, where the circuit is completed. In addition to the pumps and the heat exchangers, the system contains associated piping, valves, and connecting hoses.

The 3-in. suction connections to the reactor vessel are in the 4-in. safety injection system penetrations between the outlet hydraulic stop valves and the motor operated stop valves in coolant loops 1A and 1B. Coolant, after passing through the nonregenerative heat exchangers, can be returned to the reactor vessel either through the coolant loop with the suction connection or through the adjacent loop in that boiler chamber.

Pumps and valves for this system are on the platforms beneath the loop 1A and loop 1B boiler chamber access cubicles. All required connections are made in these shielded areas.

Design requirements. The reactor coolant temperature at the reactor outlet nozzles is estimated to be 180°F when the system is placed in operation. The coolant is reduced to this temperature by normal plant cooldown operations. The maximum temperature of the river water to the component cooling water system heat exchanger is expected to be 84°F.

Component cooling water is required only by the valve operating system air compressor, the coolant discharge and vent system pumps, and

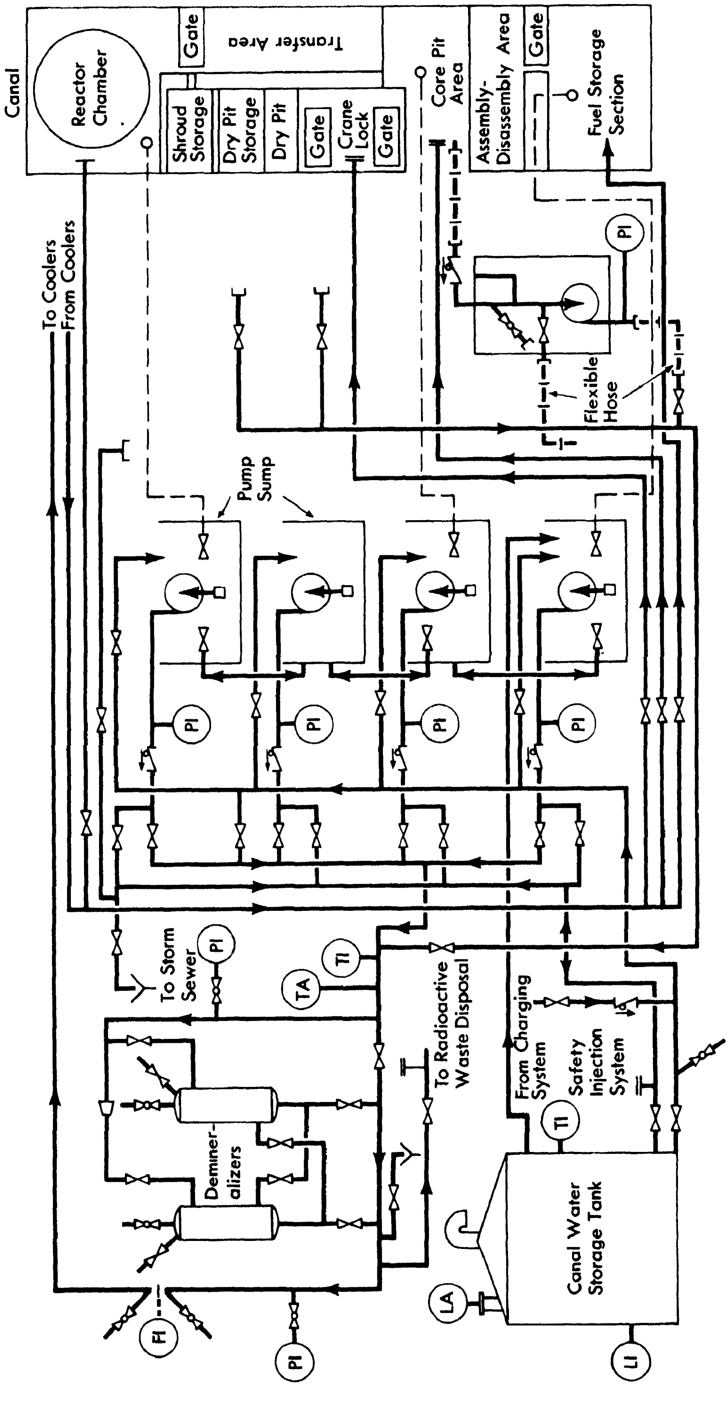


FIG. 8-7. Canal water system flow diagram.

the nonregenerative heat exchangers when the core removal cooling system is in operation. With only this equipment in operation and with the throttling valve on the cooling water side of each of the nonregenerative heat exchangers fully opened, the component cooling water flow to each nonregenerative heat exchanger is 50,000 lb/hr minimum.

The design flow rate of the reactor coolant to each nonregenerative heat exchanger during system operation is 37,500 lb/hr minimum with the motor-operated loop valves on the boiler inlet piping closed.

The time after shutdown before the system goes into operation depends on the specific core in use, power levels at which the core was operated before shutdown, and the specific shutdown procedure utilized (whether the motor-operated loop valves are closed or open).

8–1.8 Canal water system. *Function.* The canal water system (Fig. 8–7):

(1) Provides water shielding for core assemblies and subassemblies during (a) insertion into or removal from the reactor vessel; (b) transfer, assembly, and cutting; and (c) storage.

(2) Provides additional shielding above the reactor core when the core is inside the reactor vessel with the reactor head and reactor container head in place.

(3) Maintains the water used for shielding within acceptable temperature limits by providing steam heating of canal water storage tank water and removal of heat added to the canal water, including (a) heat added by the water shielded core while installed in the reactor vessel, and (b) decay heat added by the water shielded core and subassemblies while not installed in the reactor vessel.

(4) Maintains the canal water radioactivity within acceptable limits.

(5) Maintains the canal water clarity within acceptable limits by removing foreign particles such as dust and scum.

(6) Provides adequate water storage facilities to fill and drain canal sections as required for core installation or removal from the reactor, for transport of the core through the canal, for disassembly and reassembly of the core, for storage of the disassembled core, and for repair of equipment used in handling, disassembling, or assembling the core.

Design requirements. The system is manually operated except for the steam control valve in the steam line to the canal water storage tank. This valve automatically controls the inlet steam rate on the basis of tank water temperature.

The canal water must be maintained by the system to meet the following criteria: (1) sufficient quantity, (2) minimum radioactivity, (3) acceptable purity and maximum clarity, and (4) temperature within a controlled range. The means by which the system meets these requirements are described in the following paragraphs.

A storage tank is included in the system to store water required to fill a canal section or water from a canal section that is being emptied.

The canal water becomes radioactive while serving as a shield for a radioactive core and its subassemblies. Also, tools and parts that become radioactive during disassembly or assembly of the core will contribute to the radioactivity of the canal water. The radioactivity of the water is kept within tolerable limits by the two demineralizers. The reactor vessel is not opened for removal of a core until its radioactivity, temperature, and pressure are reduced to satisfactory levels. Subassemblies with damaged fuel elements will be transferred under water from the pressure vessel to the subassembly storage areas (in special containers if necessary) to prevent canal water contamination.

In addition to removing canal water radioactivity, the demineralizer beds filter insolubles from the water and, therefore, assist in keeping the canal water at acceptable purity and maximum clarity.

During disassembly or assembly, heat from irradiated subassemblies is transferred to the canal water. The temperature of the water must be controlled to avoid excessive evaporation caused by high temperature and to prevent freezing during winter. Ordinarily the air conditioning in the canal building prevents freezing. If air conditioning equipment should fail during winter, the canal water can be circulated through the steam-heated storage tank.

The canal water is normally cooled by the heat exchanger cooler. When this is not available, the water can be cooled in the component cooling water system cooler, provided that that system does not require cooling. If neither cooler is available, the canal water must be recirculated through the entire canal for the cooling effected by evaporation.

The design maximum canal water temperature is 100°F and the minimum, to provide a margin of safety above freezing, is 40°F. The maximum design temperature is coincident with a river water temperature of 80°F, the summer net heat gain, and the capabilities of the fuel-handling building air conditioning system. Although the river water temperature may exceed 80°F for short periods, the frequency with which such higher temperature will coincide with the maximum canal water temperature is expected to be low. Canal water temperatures over 100°F are allowed during emergency conditions but are not to exceed 120°F. Such temperatures would increase the evaporation of canal water to such an extent that there would be an appreciable increase in condensation inside the fuel-handling building, and corrosion rates of all metal not protectively coated would increase severalfold. Finally, the protective coating on the walls of the canal and fuel-handling building will break down at temperatures over 120°F.

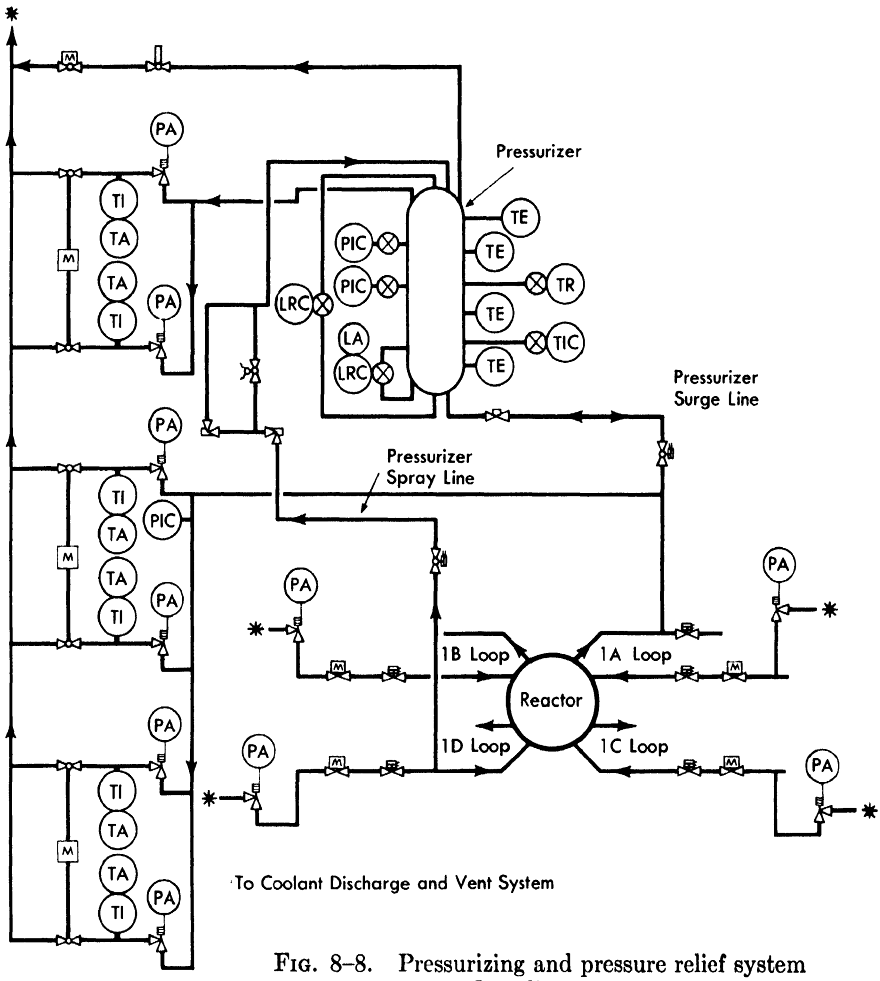


FIG. 8-8. Pressurizing and pressure relief system flow diagram.

8-2. REACTOR PLANT PROTECTION

8-2.1 Pressurizing and pressure relief system. *Function.* The pressurizing and pressure relief system (Fig. 8-8) is designed to maintain a satisfactory pressure in the reactor coolant system during steady state and transient plant operation. It must also accommodate reactor coolant volume changes which occur during temperature excursions resulting from normal plant transients; at the same time it must limit, within an allowable range, the corresponding reactor plant pressure fluctuations. The

system also prevents the pressure of the reactor coolant system from exceeding the design point, except as permitted by Section I of the ASME code and the ASA code for pressure piping: the reactor coolant piping may be subjected to pressures 20% in excess of the nominal design value during 1% of the operating life of the plant.

General description. (a) Pressure control. The pressure control portion of the system consists of a pressure vessel equipped with submerged, replaceable heater elements and a nozzle through which reactor coolant may be sprayed into the top of the vessel. The pressurizing vessel is in the auxiliary chamber of the reactor plant container and is connected to the reactor coolant system by a surge line running from the bottom of the pressurizer to the reactor outlet piping of the 1A coolant loop, inboard of the main loop hydraulic stop valve. The spray nozzle at the top of the pressurizer is connected to the reactor coolant system by a spray line to the reactor inlet piping of the 1B coolant loop, also inboard of the hydraulic stop valve.

During normal operation of the plant, a steam phase exists in the pressurizer above the free water volume. The heater elements are used to pressurize the reactor coolant system by maintaining the water in the pressurizer at the saturation temperature corresponding to the desired plant operating pressure (2000 psia). These heater elements also generate and maintain a steam phase in the dome of the pressurizer, providing a cushion for absorbing reactor coolant surges.

Plant operating pressure is maintained within prescribed limits during normal plant power level transients by expansion or compression of the steam phase in the top of the pressurizer. During power excursions the heat input to the coolant from the reactor and the heat removed from the coolant by the boiler become unequal. This changes reactor coolant volume which, in turn, varies the plant pressure. The pressurizing portion of the system is designed to limit the magnitude of these pressure variations. If the plant electrical load is suddenly increased, heat is withdrawn by the boiler faster than it is being supplied by the reactor, the reactor coolant contracts, and the water level in the pressurizer begins to drop. The plant pressure then begins to decrease, causing some of the water in the pressurizer to flash to steam. This effect, coupled with the expansion of the steam already in the dome of the pressurizer, limits the falling plant pressure, and eventually, aided by the pressurizer heaters, returns it to normal operating pressure. On the other hand, a sudden decrease in the plant electrical load causes the reactor coolant to expand, raising the pressurizer water level. The steam in the pressurizer is thus compressed and the plant pressure increases. This increase is controlled by spraying reactor coolant through the nozzle at the top of the pressurizer, thereby condensing some of the steam and lowering the pressure to within the

normal operating range. The reactor coolant spray into the top of the pressurizer is induced by the pressure drop across the reactor pressure vessel.

(b) Pressure relief. The pressure relief portion of the system protects the reactor plant when excessive positive surges result in pressures beyond the limiting capability of the pressure control portion of the system. Since isolation valves are provided for the pressurizer and the reactor vessel, self-actuated relief valves must also be provided for both vessels. Connected to the surge line on the reactor side of the isolation valves, the self-actuated reactor vessel relief valves are operated by the pressure at the outlet of the reactor vessel.

If the reactor vessel relief valves do not reseat after relieving a volume surge, an excessive amount of coolant could drain from the reactor plant with resulting serious consequences (see Chapter 11). To obviate this possibility, two pairs of self-actuated relief valves, each with a motor operated isolation valve in its discharge line, are provided for the reactor vessel. One relief valve of each pair is continuously open to the coolant discharge and vent system. The isolation valves are so interlocked that one cannot open without the other closing. A waiver permitting this installation of isolation valves for the relief valves was received from Pennsylvania Code Authorities.

Another pair of self-actuated relief valves, installed with motor operated isolation valves, interlocked as described in the preceding paragraph, is provided for the pressurizer. In addition, because of its more reliable reseating characteristics, a pilot operated steam relief valve, set at a lower pressure than the self-actuated valves, is also provided on the pressurizer to minimize the possibility of the self-actuated valves' operating. A motor operated isolation valve is also provided in the discharge line from this relief valve.

The pilot operated relief valve on the pressurizer will be the first relief valve actuated by an excessive pressure surge. If this valve fails to open, one of the self-actuated pressurizer steam relief valves will open. It is very unlikely that all three valves would fail to operate when subjected to excessive pressures; however, they are backed by the self-actuated water relief valves on the reactor vessel, set at a higher pressure and expected to operate only when the pressurizer is isolated from the reactor vessel, not as the result of an excessive pressure surge during normal operating conditions.

A self-actuated water relief valve is installed in the reactor inlet piping of each reactor coolant loop downstream of the pumps. These valves protect each loop and its components from overpressure when the loop is isolated. Pairs of relief valves installed with interlocked isolation valves similar to those on the reactor vessel are not provided because of the extreme unlikelihood that the loop relief valves will be forced to open while the loop

is in operation. No serious difficulty will result if a loop relief valve fails to reseat after it has opened while its loop was not in operation.

A thermocouple is installed in the discharge piping from each of the relief valves in the system. These may be used to detect leakage past the relief valve seats. Each relief valve also has a pressure switch actuated alarm, used to indicate a rupture of the relief valve bellows.

In addition to the pressure relief valves mentioned above, several of the auxiliary systems, e.g., the coolant purification system and the coolant sampling system, also have low capacity relief valves. Except for the relief valves in the coolant sampling system, which discharge directly to the radioactive waste disposal system surge tanks, all primary relief valves discharge to the blow-off tank in the coolant discharge and vent system. Hence, the effluent from these valves is contained and cannot contaminate the reactor plant container.

With Core I, volume surges resulting from normal load changes do not require the use of spray. However, an intermittent spray rate of 30 gpm has been provided and is available. The spray regulating valve is set to open at 2070 psia, permitting reactor coolant to be sprayed into the pressurizer steam dome. It is set to close when pressure falls to 2000 psia. A needle valve in a bypass around the intermittent spray regulating valve provides continuous normal circulation through the pressurizer to keep the surge line warm during steady state operation over extended periods of time, reducing the thermal shock to the pressurizer during positive surges and to the loops during negative surges. A flow of 0.5 gpm is required to maintain an optimum temperature gradient in the surge line from 636°F at the pressurizer to 538°F at the reactor outlet connection. Since this circulation flow is sprayed into the steam dome, it also serves to keep the internal standpipe in the pressurizer full of water. This circulation flow causes the gases in the coolant, including hydrogen, xenon, krypton, etc., to become concentrated in the vapor phase of the pressurizer dome and, at the same time, promotes identical concentrations of the solubles in the loop and in the pressurizer water.

The pressurizer is in a separately shielded compartment within the reactor plant container auxiliary chamber. The system relief valves are outside the shield walls in the 1A auxiliary chamber access cubicle, except for those on the reactor coolant loops. The manual capped valves in the spray and surge lines are operable through the 1A auxiliary chamber access cubicle shield wall. The coolant loop relief valves are in the purification cubicles in their respective boiler chambers.

Design requirements. (a) Pressure control. For normal system operation, the load range for the plant may be considered to be from 20 Mw to full power. Within this range, the electric power system dictates the rate of change of power. Consideration of the normal power range as 20 Mw to

TABLE 8-1

VOLUME SURGES RESULTING FROM LOAD TRANSIENTS

Load change, Mw	Rate, Mw/sec	Power level, Mw	Volume surge, ft^3	Approx. time to peak surge, sec
+5	0.417	20–25	−5.0	60
+10	0.417	20–30	−10.0	55
+15	0.417	20–35	−14.5	50
+20	0.417	20–40	−17.1	45
−5	0.417	40–35	+5.5	68
−10	0.417	40–30	+10.5	66
−15	0.417	40–25	+15.0	65
−20	0.417	40–20	+17.6	62

full power does not exclude operation at lower power levels. Load changes in the lower power range may be made at a rate lower than that dictated by the electric power system. During normal operation over the full power range, the power excursions which the station may be expected to accommodate are as follows:

(1) +15 Mw or −12 Mw at a step change rate.
(2) ±15 Mw at a rate of 3 Mw/sec.
(3) ±20 Mw at a rate of 0.417 Mw/sec.

The pressurizing portion of the system is designed to limit the fluctuation of reactor coolant system pressure within the range from 1850 psia to 2180 psia during the power excursions listed above.

The maximum permissible upper pressure limit of 2180 psia was selected to be compatible with the core pressure drop and with the settings and tolerances of the primary relief valves. The lower pressure limit of 1850 psia was arbitrarily selected to prevent the reactor plant pressure from dropping below the saturation pressure at the reactor hot spot, thereby minimizing the possibility of local boiling occurring there.

Table 8–1 lists the volume surges calculated to occur as the result of 0.417 Mw/sec load transients in the lower power range. Simulator studies have substantiated the assumption that the magnitude of the volume surge varies directly with the magnitude and inversely with the rate of the load transient over the applicable range. Also, the magnitude of the volume change accompanying a given load change in the low power range is higher than that for an identical load change at a higher power range. Hence, the maximum surges listed are the largest normally expected to occur during Core I operation. For Core I, the pressurizing and pressure relief system

can accommodate these transients, maintaining the reactor plant pressure between 1850 psia and 2180 psia without the aid of sprays.

The surges listed are based on a high flow core with a temperature coefficient of $-2 \times 10^{-4}\ \Delta k/°F$ and a total reactor coolant active volume of 2490 ft^3. Greater volume changes would be expected for comparable load transients when a low flow core is used.

In addition to the pressurizer water storage requirements imposed to satisfy the negative volume surges corresponding to design load transients, the pressurizer must also contain sufficient water to permit, without emptying, the contraction of the reactor coolant which will occur during the first two hours of decay heat removal. However, since the pressurizer water volume necessary to satisfy this requirement is less than the volume needed to maintain suitable reactor plant pressures during design load transients, this requirement was of little importance in sizing the pressurizer.

During plant start-up, the rate of temperature increase in the pressurizer must not exceed 200°F per hour, to hold thermal stresses in the vessel within acceptable limits.

The heating elements in the pressurizing vessel are divided into three groups, each performing a distinct function. The first group of heaters is operated continuously to make up the normal heat losses from the system. The second is cycled on and off continuously to maintain normal operating conditions in the reactor plant. The third is required for initial plant warmup. A portion of this third group is also used to restore the required pressurizer temperature after a surge; the remainder is normally inoperative during plant operation. All pressurizer heaters have both low water level and high temperature cutoff switches.

(b) Determination of pressurizer size. The pressurizer is designed to prevent coolant pressure (as a result of surges caused by design plant load changes) from falling below 1850 psia or rising above 2180 psia either during or after the load transient.

The allowable positive pressure excursion of 180 psi above the nominal plant operating pressure of 2000 psia is based on preventing the plant pressure from exceeding a point that will unseat relief valves during a design load change; thus flexibility of plant operation within design limitations is permitted. The lowest relief valve set point, that of the pressurizer pilot operated relief valve, is fixed at 2190 psia. Since the tolerance on the set pressure for this valve is ± 10 psi, the highest system pressure that will not cause a relief valve operation is 2180 psia. Hence, if the pressurizer is to prevent relief valve operation resulting from a design load change, it must be sized to maintain the pressure peaks caused by the design load changes within the 2180 psia limit.

The maximum allowable negative pressure excursion of 150 psi below the nominal plant operating pressure is a nominal choice based on operating

experience with similar plants, which indicates that such a limit will prevent local boiling at the reactor hot spot. The actual maximum pressure decrease accompanying the largest negative volume surge (-17.7 ft^3) will be considerably less than the allowable limit of 150 psi. It will be approximately 100 psi, depending on the relative volumes of steam and water in the pressurizer prior to the negative surge.

So that the pressurizer would be sized to satisfy the design limitations discussed above, it was subdivided into a number of distinct volumes, each of which was sized to meet certain requirements. The total tank volume consists of the following seven individual volumes, each sized in the manner described below:

(c) Volume 1—bottom head and heater volume. The accepted vendor design provides a total of approximately 63 ft^3 of water surrounding the heaters and in the bottom head of the pressurizer. The water level corresponding to this volume is just above the top row of heaters.

(d) Volume 2—level instrument error volume. Two level measuring instruments are provided on the pressurizer. One is a short scale instrument (125 in.) to be used during operation; the other is a wide scale instrument (190 in.) to be used during plant start-up and shutdown. The short scale instrument is quite accurate in the normal operating range (inherent error of $\pm 2\%$) and is calibrated for normal operating pressure and temperature. The wide range instrument can be manually compensated for density changes throughout a varying pressure and temperature range. Consequently, its error is also approximately $\pm 2\%$, providing there is continuous compensation. Assuming the most conservative case, the maximum possible error in pressurizer water level indication during normal operation is 2% of 125 in., or 2.5 in. Since the inside diameter of the pressurizer is such that 1 in. of water level represents a volume of 1.3 ft^3, the maximum water volume which must be provided to allow for variations due to instrument error is 2.5 in. $\times$ 1.3 ft^3/in., or 3.3 ft^3.

(e) Volume 3—flashing water volume. A negative volume surge results in an expansion of the steam initially in the pressurizer and a corresponding decrease in pressure of the saturated water volume. This pressure decrease will cause a certain amount of the ballast water in the vessel to flash to steam. Based on a maximum initial water volume of 115 ft^3 and a maximum pressure decrease of 150 psi, and assuming that the expansion process follows the saturation line, the greatest volume of water which will flash to steam is 3.5 ft^3. This volume, then, must also be provided in the pressurizer vessel.

(f) Volume 4—negative surge volume. The load change resulting in the worst negative volume surge is the four-loop, full flow design load change from 20 Mw to 40 Mw at the rate of 25 Mw/min with only temperature coefficient control. The temperature coefficient of reactivity is taken as

$-2 \times 10^{-4}\ \Delta k/°F$. Simulator results indicate that the above load change causes the average loop temperature to drop 6°F in 60 sec. The negative volume surge of 17.7 ft^3 may be calculated in the following manner.

The four loop active plant water volume, that volume affected by temperature transients, is 2490 ft^3. Before any load change, the reactor coolant system conditions are nominally 2000 psia and 525°F. The active system water weight is then given by

$$W = \frac{V}{v_1},$$

where W = system active water weight, V = system active water volume, v_1 = initial specific volume, and W = 2490 ft^3/(0.02074 ft^3/lb) = 120,050 lb.

At the end of the transient, when the average loop temperature has dropped 6°F to 519°F, the reactor coolant system water weight will be given by

$$W = \frac{V}{v_2},$$

where v_2 = final specific volume, W = 2490 ft^3/(0.0206 ft^3/lb) = 120,750 lb.

The increase in the loop water weight is 700 lb. This water must be made up by the pressurizer, whose average temperature during the transient is 631°F. The water volume required in the pressurizer to make up this surge is 700 lb $\times$ 0.0253 ft^3/lb, the specific volume of the water in the pressurizer, or 17.7 ft^3.

Calculations have indicated that during this surge the pressure will drop to a low point of approximately 1890 psia, which is sufficiently above the maximum allowable low pressure point of 1850 psia.

(g) Volume 5—power level variation volume. The temperature distribution throughout the reactor coolant loops at 0% power is different from the distribution corresponding to 100% steady power, even though the average loop temperature is not increased. The additional water volume of 10 ft^3 required by this change in loop temperature distribution is provided in the pressurizer, so that at 100% power the steam space is sufficient to accommodate the design positive surges.

(h) Volume 6—volume change due to T_{av} variation. The maximum allowable variation in average loop temperature is ±3°F. This deviation from 525°F is due to the temperature control "dead band." The water volume affected by this long term change in temperature is 2490 ft^3. A water volume of 18 ft^3 is included in the pressurizer to avoid charging or bleeding any water that might otherwise be required as the result of normal variations in plant temperature.

(i) Volume 7—steam volume. The pressurizer volume remaining after all the above volumes have been filled with water will be occupied by steam. This steam volume must be sufficient to accommodate the maximum volume surges resulting from the design load transients listed in the system description, while still limiting the accompanying pressure excursion to 2180 psia or less. To determine the steam volume required, the following assumptions must be made:

(1) The initial condition of the steam is dry and saturated at 2000 psia and 636°F.

(2) The steam phase compression process follows the saturation line in the direction of increasing pressure, whether or not a coolant spray is being injected into the steam.

(3) Any spray flow that is provided in the pressurizer — to add conservatism to the design and to provide for the possibility of larger surges resulting from future cores — is 100% effective, and its temperature is raised to at least that of the steam.

Calculations have indicated that the greatest positive volume surge, 17.6 ft^3, will occur as the result of a plant load change from 40 to 20 Mw at the rate of 25 Mw/min. A family of curves can be plotted for various spray rates, from which the initial steam volume required to accommodate various positive surges without allowing the plant pressure to exceed 2180 psia can be determined. Since the volume of the pressurizer was sized at 261 ft^3 on the basis of an isentropic steam compression process and the maximum water volume required in the pressurizer is 115.5 ft^3, the minimum steam volume in the pressurizer prior to a positive surge will be 145.5 ft^3.

To summarize, the over-all volume of the pressurizer includes the following:

Water volume	
Bottom head and heater volume	63.0 ft^3
Level instrument error volume	3.3 ft^3
Flashing water volume	3.5 ft^3
Negative surge volume	17.7 ft^3
Power level variation volume	10.0 ft^3
T_{av} variation volume	18.0 ft^3
	115.5 ft^3
Steam volume	
Positive surge volume	17.6 ft^3
Balance	127.9 ft^3
	261.0 ft^3

The inside diameter of the pressurizer is 54 in., the over-all height (excluding attachments), 18 ft $4\frac{3}{4}$ in.

(j) Determination of required heater capacity. A period of ten minutes was selected as a reasonable time for plant pressure to return to the normal steady state operating value of 2000 psia, following a design load increase. The worst positive load transient was selected as the design condition, since this represents the case for which the maximum amount of heat which must be made up by the pressurizer heaters is withdrawn from the reactor coolant. During a negative load transient, the rate of heat withdrawal from the reactor coolant decreases, thus decreasing the load demand on the pressurizer heaters.

In estimating the required heater capacity, the following factors were considered:

(1) A negative surge from the pressurizer results from a power increase transient. The surge represents the contraction of the reactor coolant as it yields some of its thermal capacity during the transient. As equilibrium is approached again, the reactor plant returns the negative surge volume, now at T_h, to the pressurizer.

(2) In addition to the return of the negative surge volume to the pressurizer at a reduced temperature, the pressurizer water volume is increased by the amount due to the increase in the power level of the plant. This increase in volume also comes into the pressurizer at temperature T_h.

(3) As the initial negative surge progresses, the pressurizer pressure and temperature decrease and some pressurizer water flashes, tending to keep the entire water and steam mass in the pressurizer saturated.

(4) The pressurizer tank itself, and all the metal originally in contact with water or steam at 636°F, lose heat to the lower temperature pressurizer water and steam as the negative surge, which precedes the eventual positive surge, progresses.

The above transient heat losses must be made up by the pressurizer heaters, since they are the only source of heat energy provided to maintain the pressurizer at 636°F.

(k) Pressure relief. The pressure relief system is designed to protect the reactor plant from overpressure during the most severe operating emergency, which assumes simultaneous loss of power demand, loss of coolant pump power, and failure of the reactivity control system. A nuclear accident during an attempted startup of the reactor at low temperatures with all reactor coolant loop isolation valves closed has not been considered in the system design because such an attempt is extremely unlikely.

A self-actuated relief valve for each reactor coolant loop protects the loop when cold from overpressure due to any of the following causes:

(1) Improper control of the bleed-feed operation during loop warmup.

(2) Steam leakage from an operating boiler to a down boiler.

(3) Accidental filling of an empty down boiler with hot feedwater while the associated loop is isolated and pressurized.

(4) Overcharging an isolated loop with a 3000-psi charging pump.

Each coolant loop relief valve is capable of relieving water at the rate of 50 gpm, the total flow rate resulting from the use of both charging pumps at full speed for supplying water to an isolated loop. This is the most severe of the conditions listed above.

The pilot operated steam relief valve on the pressurizer can relieve steam or water at the rate of 1.6 ft^3/sec. This is the surge rate resulting from simultaneous loss of power demand and failure of the reactivity control system and/or the reactor power and temperature control system to insert the control rods. The reactor will subsequently be shut down by the effect of its negative temperature coefficient alone. For Core I, the surge rate due to an accident of this type has been determined by simulator studies to be somewhat less than 1.6 ft^3/sec.

Each of the self-actuated steam safety valves on the pressurizer and the self-actuated water relief valves on the reactor vessel is sized to relieve at a rate of 3.2 ft^3/sec. This is the surge rate during the most severe operating emergency (simultaneous loss of power demand, loss of coolant pump power, and failure of the reactivity control system and/or the reactor power and temperature control system to insert control rods). It is also assumed that the pilot operated pressurizer relief valve has failed to open at this time. Subsequent reactor shutdown will be accomplished by the effect of its negative temperature coefficient. The surge rate for Core I resulting from an accident of this type has been determined by simulator studies to be 1.75 ft^3/sec.

The pressure settings of the system safety and relief valves are based upon a reactor vessel outlet pressure of 2000 psia and a system design pressure of 2500 psig.

In general, the piping, valves, and other components of the pressurizing and pressure relief system are designed for pressures up to 2500 psig and temperatures up to 675°F, and comply with the requirements of Section I of the ASME Boiler Code. In addition, all metal surfaces which may be exposed to reactor coolant are of type-304 stainless steel or stellite.

(l) Determination of relief valve settings. In determining the safety and relief valve settings for the pressure relief system, the presently attainable tolerance for blowing has been utilized for the self-actuated steam and water relief valves. The weep allowance (above which leakage begins) is only 25 psi, since weep will be tolerated if an emergency necessitating relief valve operation should occur. Presently attainable tolerances for blowing have been utilized for the pilot operated steam relief valve on the pressurizer. Weep is not a problem for this valve. The following settings and allowances have been determined:

	Pressure (psig)
Maximum permissible pressure	2650
Pressure drop through reactor pressure vessel (RPV)	50
RPV self-actuated water relief valve accumulation (6%)	150
2nd RPV self-actuated water relief valve setting	2450
1st RPV self-actuated water relief valve setting	2450
Blow tolerance	50
Minimum blowing pressure	2400
Weep tolerances	25
Minimum weeping pressure	2375
Maximum permissible RPV outlet or pressurizer pressure	2375
Pressurizer self-actuated steam relief valve accumulation (3%)	65
Blow tolerance	50
Pressurizer self-actuated steam relief valve setting	2260
Blow tolerance	50
Minimum blow pressure	2210
Weep tolerance	25
Minimum weeping pressure	2185
Blow tolerance	10
Pressurizer pilot operated steam relief valve setting	2175
Blow tolerance	10
Minimum blowing pressure	2165
Weep tolerance	0
Minimum weeping pressure	2165
RPV outlet pressure	1985
Permissible pressure surge to weep point	180

The settings of the reactor coolant loop relief valves were determined as follows: The ASME Code, Section I, permits the 18-in. piping to be overstressed for short periods by 20% of the design pressure. During these periods, the maximum loop pressure may, then, be 1.2 x 2500, or 3000 psig. The relief valves must, then, be set low enough so that at maximum accumulation the pressure does not exceed that value. The maximum accumution expected in relief valves of the type used in this application is 6% of the relief valve setting, making the set point 3000/1.06, or 2830 psig.

8–2.2 Decay heat removal system. *Function.* The primary function of the decay heat removal system is to dissipate reactor decay heat fast

enough to protect the core from damage following an emergency shutdown due to electrical power failure to the reactor coolant pumps. The system will ensure the integrity of the plant during such an emergency.

As an adjunct to this primary requirement, the decay heat removal system can be used to remove reactor heat during plant shutdowns or during plant operation at low reactor power levels. Although the major function of the system is, of necessity, performed without using electrical power, the auxiliary function can be accomplished with plant power available.

General description. The system consists primarily of a steam relief valve set to operate at 707 psia and sized to dissipate 7000 kw of heat. It is installed in the turbine room basement on the 12-in. steam line from the 1B reactor coolant loop steam generator, and is normally isolated from the steam header by a motor operated gate valve. However, the isolation valve opens automatically upon failure of the AC power supply to all the reactor coolant pumps. During relief valve operation, reactor coolant is induced to flow by natural convection through the reactor coolant loops. In this manner, decay heat is removed from the reactor, transferred across the steam generators to the secondary portion of the plant, and discharged to atmosphere through the decay heat steam relief valve.

To accomplish the auxiliary functions mentioned above, the isolation valve for the steam relief valve can be remotely operated from the main control room.

Since reactor decay heat is dissipated by discharging steam from the secondary portion of the plant to the atmosphere, make-up water will eventually be required for the steam generators. In an emergency involving loss of all electrical power to the reactor coolant pumps, it must be assumed that power is also lost to the boiler feedwater pumps. To enable make-up a charging pump must be used, since it can draw power from the emergency diesel generator. Water is directed from the coolant charging system fill header through a crossover line to the discharge header of the boiler feedwater pumps. A relief valve is installed on this line to prevent inadvertent overpressuring of the boiler feedwater piping. To permit maintenance of the relief valve, a stop valve is installed in the crossover line on the feedwater side of the relief valve.

Design requirements. Reactor decay heat removal requirements are based on a plant electrical output of 100 Mw for 600 hr before the emergency occurs.

The reactor plant must be capable of sustaining, without damage, casualties interrupting normal heat removal schemes, including a complete loss of AC electrical power, for:

(1) Any time up to 2 hr without requiring action by plant operating personnel.

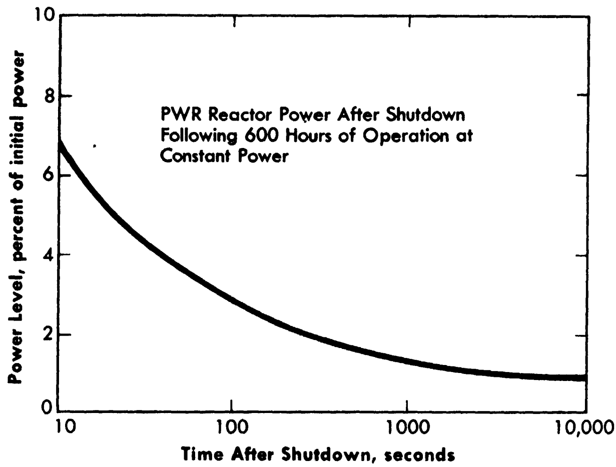


FIG. 8–9. Decay heat curve.

(2) Any time up to 8 hr with corrective action by operating personnel, using only station facilities.

(3) Indefinitely, with corrective action within 8 hr by operating personnel and the availability of power or facilities external to the station power or facilities.

Corrective action by plant operating personnel must not require entering the plant container.

Regardless of the type of casualty and independent of the availability of personnel, the reactor will not release fission products outside the reactor coolant systems so long as the reactor plant integrity has not been violated.

The system is capable of auxiliary functions such as the removal of decay heat after plant shutdown or during operation at lower power levels. The design of the system was based on the PWR decay heat curve shown in Fig. 8–9.

System operation. Immediately following the loss of AC power, the turbine throttle and the auxiliary steam valves close, and the reactor control rods are dropped into the core. The reactor plant temperature then begins to rise as a result of the relatively large amount of heat still being generated by the reactor and of the sudden decrease in the coolant flow rate caused by pump coastdown, which, in turn, decreases the amount of heat being transferred in the boilers. During the first few seconds after an accident, much of the decay heat is absorbed by the reactor coolant and the plant metal or is lost by radiation. The remainder is carried to the boiler heat exchangers, where it is transferred to the water on the secondary side of the boilers. The transfer of this heat to the secondary water reduces

the temperature of the reactor coolant in the steam generators, thereby causing a density difference between the coolant at the boiler inlet and that at the boiler outlet. As a result of this density difference and the elevational difference between the boilers and the reactor, natural circulation of the coolant through the loops is established, and operation becomes continuous. The heat transferred to the secondary water increases the pressure of the steam in the boiler steam drums and interconnecting piping until it reaches the set pressure of the steam relief valve. At this time, the relief valve opens and steam is blown to the atmosphere, thus removing heat from the plant. In effect, the boilers act as heat sinks to absorb the reactor decay heat.

The operation of the boilers as heat sinks, in conjunction with the steam relief valve discharge, limits the temperature rise of the reactor coolant. In turn, coolant expansion is also limited to such a degree that the surge volume will not cause the pressurizer pilot operated relief valve to open. The set pressure of the steam relief valve is based on calculations which include a demonstration that the limiting temperature, as described above, is not reached in the reactor coolant system before the steam relief valve begins to lift.

The decay heat removal system has been designed so that no corrective action by plant personnel is required for at least two hours. However, the emergency diesel generator should be placed in service as soon as possible to operate a charging pump for adding water to the steam drums and the pressurizer when necessary.

Reactor coolant grade water from the reactor plant water storage tank can be added to the boilers during a loss of AC power by utilizing a charging pump (which can be powered from the emergency diesel generator) and the crossover line between the coolant charging system and the boiler feedwater system. When the valving is properly aligned and the charging pump is operating, reactor coolant grade water passes through the boiler feedwater regulating valves and into the boiler steam drums. Should enough water be pumped into the boilers to cause the feedwater regulating valves to close, the charging pump discharge pressure increases to approximately 1200 psig, and the water relief valve then opens to prevent excessive pressure in the boiler feed piping. However, by careful operation of the charging pump, it is possible to prevent the water relief valve from lifting. The feedwater regulating valves are operable at this time, since they can also draw their power from the emergency diesel generator.

The reactor continues to generate decay heat at a decreasing rate during the emergency; eventually the heat losses by radiation become equal to the decay heat rate. Then coolant temperature and volume decrease. The steam relief valve is sized to prevent the water level in the pressurizer from dropping below the zero point of the level indicator. Unless a water level is

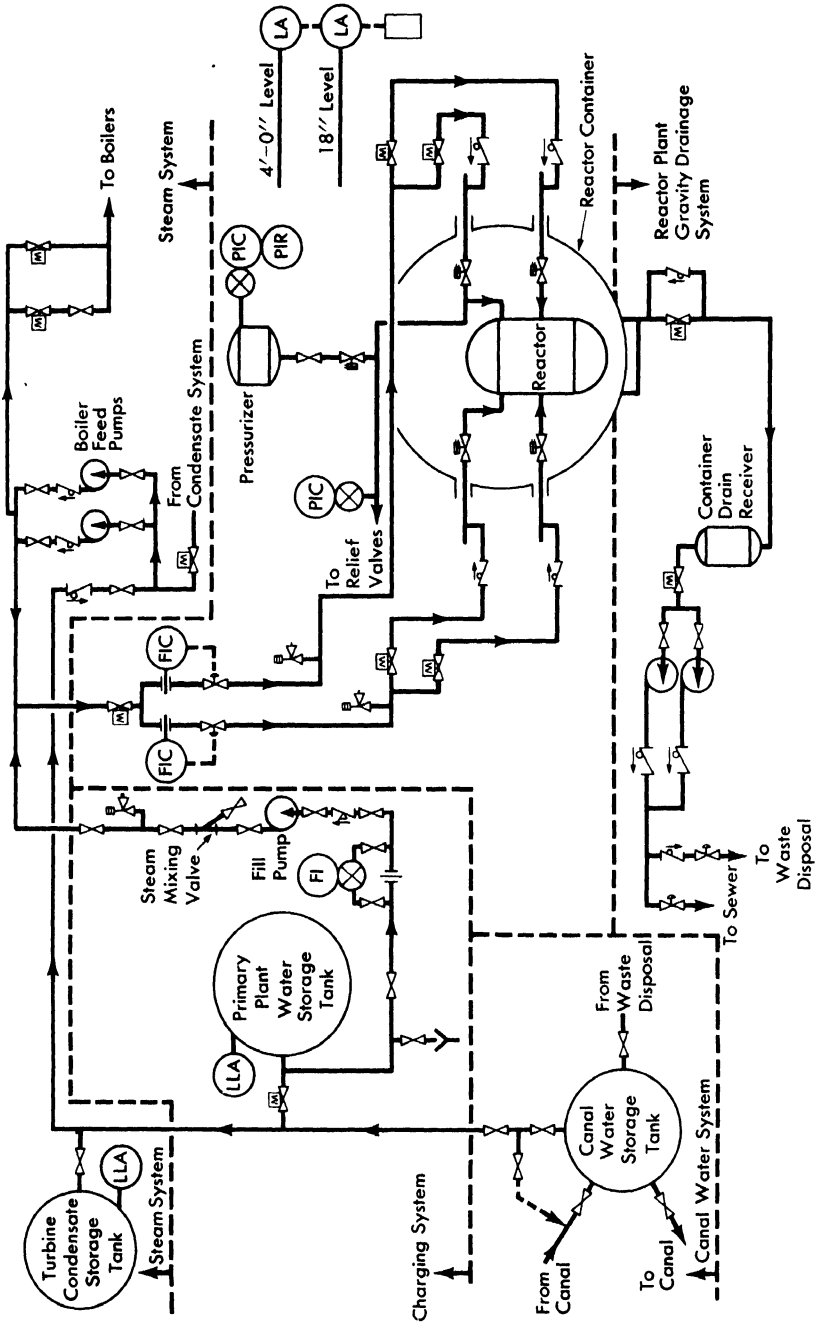


FIG. 8–10. Safety injection system flow diagram. Portions of other interconnected systems are also shown.

maintained in the pressurizer, there can be no natural convection circulation of coolant, and boiling can occur in the reactor. Should it be desired to raise the pressurizer water level, a charging pump can be used in the manner normally incorporated for reactor coolant system charging.

The decay heat removal system can be used to remove decay heat following a normal station shutdown in which complete cooldown is not necessary or desirable, maintaining the reactor at low power levels with no turbine demand. Such a condition exists when the station is shut down for an overnight capacity reduction. This method of operation is utilized with normal AC power available at the station.

8–2.3 Safety injection system. *Function.* The safety injection system (Fig. 8–10) provides the means for injecting water into the reactor pressure vessel following any accident resulting in irrecoverable loss of coolant. Cooling water must be supplied to the reactor core to prevent (1) extensive core meltdown and (2) a zirconium-water or zirconium-steam reaction, which can occur if core decay heat raises core temperature enough to initiate such a reaction. For such an accident, the rupture must be of such magnitude as to exceed the flow capacity of two charging pumps and as to be beyond the scope of any other emergency procedure.

General description. The safety injection system consists of valves and piping which connect the reactor plant water storage facilities to the reactor coolant system. In addition to the equipment specifically assigned to the safety injection system, the system utilizes several components and portions of other systems throughout the reactor plant. System operation can be divided into two basic phases. The first, largely automatic, provides the means for injecting large volumes of water into the reactor coolant system and, through the rupture, into the container. The second provides for circulating and cooling the large volume of water that has poured into the container.

(a) First phase operation. When operation of the system is initiated, the discharge of the boiler feed pumps is automatically diverted from the feedwater piping into the safety injection piping leading into the container. A preset flow controller insures equal flow into two safety injection headers. In the auxiliary chamber, each header is again divided, providing four safety injection outlets. These lines lead into the boiler chambers to penetrate the four reactor coolant system reactor outlet lines. This provides practically equal safety injection flow into each reactor outlet line at a point near the reactor. This water will fill the reactor vessel and chamber or will keep the core covered, depending on the location and magnitude of the rupture.

The feedwater supply to the boiler feed pumps is automatically switched when system operation is initiated; the pumps draw directly from the

turbine condensate storage tank rather than from the turbine condenser. Operator action is required to open the reactor plant water storage tank and the canal water storage tank outlets, so that the water supply in all three storage tanks can be pumped into the reactor plant.

The reactor plant container gravity drainage system is interlocked with the safety injection system to provide automatic closure of the motor operated valves in the reactor chamber drain line and the reactor plant container drainage tank outlet line. This allows water to flow from either the boiler chambers or the auxiliary chamber into the reactor chamber and, at the same time, retains any spilled water in the container.

(b) Second phase operation. With a severe rupture, the boiler feed pumps will empty the available water storage facilities in several minutes. The boiler feed pumps then stop, and a cooling circuit recirculates the water which floods the container.

The reactor plant container gravity drainage system is started to pump water from the container to the surge tanks of the radioactive waste disposal system. This water is then pumped into the canal water storage tank. The water is then cooled in the canal water cooler and returned to the reactor plant water storage tank. The charging system fill pump pumps water from this tank into the safety injection headers via the decay heat removal system crossover line between the charging header and the boiler feedwater system. This operation is continued until reactor decay heat is reduced sufficiently to begin recovery operations.

Design requirements. (a) Safety injection initiation. System action is initiated when (1) the safety injection control switch on the reactor console is tripped, and (2) either the pressurizer pressure or reactor vessel pressure drops to 500 psig. Both (1) and (2) must be satisfied before the system will operate.

(b) Injection flow rate. With both boiler feed pumps in service, the safety injection flow controllers are set for 1500 gpm, providing a total water flow of 3000 gpm to the reactor if operation of the system is initiated. If only one boiler feed pump is in service, the flow controllers are set for 750 gpm, providing a total availability of 1500 gpm. In the event of a large break below the level of the core (reactor inlet piping in reactor chamber) with four coolant loops in operation, a flow rate of 3000 or 1500 gpm enters the top of the reactor vessel at four equally spaced points. For ruptures anywhere but in the bottom inlet piping, at least 3/4 flow is supplied to the vessel, giving a minimum supply of 2250 or 1125 gpm to the vessel. With three coolant loops and two boiler feed pumps in operation, a minimum of 1500 gpm will be supplied to the vessel for breaks below the level of the core; with three loops and one boiler feed pump, a minimum of 750 gpm.

Two facts are pertinent:

(1) Both boiler feed pumps will normally be available; reactor operation at full power *requires* that both pumps be operating.

(2) To be below the core, a system rupture must be in the lower part of the reactor vessel itself or in the four short sections of 18-in. pipe immediately adjacent to the vessel. All other parts of the system are effectively "above the core" because of the piping layout.

(c) Maximum injection. The safety injection system must supply enough water to insure that the core is covered, regardless of the location and magnitude of the rupture. However, the structural load limits of the container chambers must not be exceeded by the weight of the large volume of water that may issue from the rupture.

(d) Injection water storage requirements. A total water storage capacity of 88,000 gals must be available at all times during reactor plant operation in case the safety injection system is required. This requirement may be relaxed only when the condition of the plant is such that reactor decay heat would be insufficient to raise an uncovered core to melting temperatures. This storage requirement is divided among the turbine condensate storage tank, the reactor plant water storage tank, and the canal water storage tank. Each tank is provided with a low level alarm contact set at the minimum level, which actuates an annunciator in the main control room.

(e) Point of injection. One safety injection line penetrates each reactor coolant loop at a point in the reactor outlet line between the hydraulic stop valve and the motor operated stop valve. The penetration to the 18-in. pipe is in each boiler chamber.

(f) Coolant level alarms. Three alarm contacts in the containers actuate lights in the main control room, so that the operator can determine the approximate level of water in the container and whether the rupture is outside or inside the reactor chamber. One contact is actuated when the level in the reactor chamber reaches the upper lip of the interconnection between the reactor chamber and the auxiliary chamber. The other two contacts are actuated when the level in the auxiliary chamber reaches $1\frac{1}{2}$ and 4-ft levels, respectively. Indicating lights in the control room are lit when the appropriate level contact is made. The three signals are connected to the annunciator so that actuation of the first contact will sound an alarm.

System performance analysis. An extensive analysis was made to determine whether the safety injection system, as designed, can prevent core meltdown and significant chemical reaction following a loss-of-coolant accident. The criteria used in this study were the results of the tests and calculations. This work is reported in WAPD-SC-543 and -544.*

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*"Zirconium-Water Reaction Data and Application to PWR Loss-of-Coolant Accident" and "PWR Loss-of-Coolant Accident — Core Meltdown Calculations," respectively. (Available from Office of Technical Services, U. S. Department of Commerce, Washington 25, D. C.).

One objective of the analysis was to determine whether the rates and method of injection of cooling water, as provided by the safety injection system, were sufficient to prevent such meltdown and chemical reaction. The loss-of-coolant case considered in the analysis involved an extreme assumption with respect to the size of the break. The following conditions, among others, were assumed:

(1) A break equivalent in area to two 15-in. diameter holes, resulting in uncovering the core in 15 sec.

(2) A total reactor output of 940×10^6 Btu/hr.

(3) A core history such that the decay heat follows the 600-hr curve from an initial power of 940×10^6 Btu/hr.

For an accident occurring under these conditions, the minimum safety injection system flow rate required to cool various regions of the core was determined. In addition, for a system water flow rate that will prevent the core from reaching melting temperature, determination was made of the maximum permissible elapsed time between reactor scram and the first introduction of system water.

(a) Ruptures above the core. The safety injection system can start to operate when the reactor coolant system pressure has dropped to 500 psi, or about 15 sec after rupture of an 18-in. pipe. Assuming that system flow initiates at this time, the system can prevent core meltdown as a result of a major coolant loop break by quickly re-covering the core, provided the break is above the level of the reactor core. The times required at various pumping rates to re-cover the core to its midline, following initiation of operation of the system for a break occurring above the level of the core, are 212 sec at 750 gpm, 106 sec at 1500 gpm, 71 sec at 2250 gpm, and 53 sec at 3000 gpm. All these recovery times would completely prevent any meltdown or zirconium-water reaction.

(b) Medium-sized ruptures below the core. When 1500 gpm flow reaches the reactor vessel, it is calculated that the core would be quickly re-covered and maintained completely covered with a pipe break as large as 4 in. in diameter, even if the break were below the level of the core. If 3000 gpm should reach the reactor vessel, the core would be kept covered with water with a break as large as 6 in. in diameter occurring below the core. Although this flow rate must eventually be reduced to prevent exceeding the structural load limit of the container, the water level in the reactor chamber would, by that time, be within a few inches of the top of the core. After this time, the 1500-gpm boiler feed pumps would be shut down, and the 200-gpm fill pump would be used to supply water for removal of decay heat.

It is concluded that the 3000-gpm safety injection system can keep the core covered for openings in the reactor plant equivalent to 6 in. in diameter, even if located below the core. No core meltdown or release of fission

products from within the core is expected as long as it is covered with water. At a system flow rate of 1500 gpm, some meltdown is expected.

(c) Large ruptures below the core. In case of a break below core level that is larger than that equivalent to a pipe 6 in. in diameter, the safety injection system would not be able to keep the core covered until the water level in the reactor chamber had become high enough to cover the core.

In the unlikely event that such an accident should occur, it would be necessary to inject water until the level in the spherical reactor chamber and in the reactor pressure vessel immersed the core. At the maximum water level in the reactor chamber, when water would overflow through the 12-ft diameter interconnecting duct to the auxiliary chamber, the water level within the pressure vessel would be about eight inches from the top of the active core. The times required at various pumping rates to re-cover the core to its midline, following initiation of system operation with a break occurring below the level of the core, are 84 min at 750 gpm, 42 min at 1500 gpm, 28 min at 2250 gpm, and 21 min at 3000 gpm.

During the time required to immerse the core, water and steam would flow through the fuel assemblies, thus removing decay heat. The effectiveness of the system in immersing and cooling the core depends upon the delay time in initiating system operation, and upon the rate at which the system supplies water directly to the core and the reactor container. Ideally, the water supplied by the system would be distributed radially to the fuel assemblies in proportion to the decay power generated in each assembly. As the system is now designed, safety injection water is supplied to the fuel assemblies with the highest decay heat fluxes, i.e., the assemblies in the seed region and the adjacent blanket assemblies. The safety injection water is directed to the control rod shrouds. Most of this water will run down through the seed, with the remainder splashing away from the shrouds to the adjacent blanket assemblies (designated as Regions 2 and 3 of the blanket). While the splashing of water from the shrouds above the seed will supply water to some of the remaining blanket assemblies (designated as Regions 1 and 4 of the blanket), it is not certain that all of these assemblies will receive cooling water. Therefore, it was assumed for the analysis of a large break below the level of the core that no water splashes to Regions 1 and 4, but that these regions receive some steam (in proportion to the flow distribution) and that some cooling results.

To determine the effectiveness of core cooling when the safety injection system is in operation, calculations of core temperature excursions were made, assuming the conditions of break size and location, reactor output, and core history listed earlier in this discussion. The calculations take into account the heat of reaction of zirconium with water and the heat removed from the fuel elements by steam convection. Studies of the high flux portions of the fuel elements in the seed and each blanket region

indicate the allowable delay times for various rates of steam supply to cool the fuel elements. The seed is the most critical region, with a time delay of 89.8 sec allowable to prevent melting when the injection rate of water is 250 gpm. Lower rates, down to 50 gpm, would suffice to cool the seed and Regions 2 and 3 of the blanket if operation of the injection system were started earlier.

The analysis shows that the 3000-gpm maximum capacity of the safety injection system, supplied for cooling of the core after a worst case loss-of-coolant accident (equivalent in area to two 15-in.-diameter holes below the core), is in excess of the amount needed to prevent fission product release. Because of difficulties in obtaining proper distribution, this excess is desirable to assure that more than the minimum will be supplied to the seed and to blanket Regions 2 and 3. However, since injection water is not distributed directly to Regions 1 and 4, it is desirable to use this injection rate to immerse the core as soon as possible.

With 3000 gpm supplied, immersion of the core to within 8 in. of the top is expected within 25 min. The level is expected to reach the midline of the core in 21 min, so that some cooling would be expected at that time for blanket Regions 1 and 4. Since the maximum flux rods in blanket Region 1 are expected to begin melting in 19 min, some reaction and melting could be expected here. At the most, this is not expected to exceed about 2½% of the Zircaloy-2 cladding in Region 1 of the blanket if all the blanket rods in the system were equivalent to the rod for which computations were made. Since this is not the case, a much lower amount of reaction would be expected. No melting would be expected in Region 4, since this region is immersed before melting temperature is reached.

With one boiler feed pump available, 1500 gpm will be supplied to the core, and the level will reach the midline of the core about 42 min after the system begins operating. Under these circumstances, some meltdown of cladding will probably take place in Regions 1 and 4 (approximately 57% and 36%, respectively, with release of fission products). However, all other regions of the core are expected to be amply cooled.

Finally, it should be recognized that many of the conditions assumed in the analysis were conservative, i.e., led to results more severe than can be expected even in the postulated loss-of-coolant accident. Thus, the safety injection system, which at the minimum meets the requirements imposed by these results, is considered to have sufficient inherent safety factors to limit core meltdown and chemical reaction.

8–3. Reactor Plant Information

8–3.1 Reactor coolant sampling system. *Function.* The reactor coolant sampling system has facilities for obtaining cold (approximately 120°F) samples of reactor coolant from the influent and effluent sides of each

coolant purification system demineralizer and for analyzing the samples chemically, physically, and radiochemically. In addition, the system has facilities for collecting samples of soluble and insoluble loop corrosion and fission products for analysis elsewhere.

Information provided by this system is used in determining reactor coolant chemical conditions during testing, initial plant startup, and all subsequent plant operation. The data are also utilized to estimate corrosion of the reactor coolant system, predict performance of the coolant purification system, estimate fission product activity in the reactor coolant, and provide other basic reactor plant water chemistry data.

In addition, the system is convenient for introducing hydrogen into the reactor coolant system.

General description. The system (Fig. 8-11) obtains samples from four locations in the coolant purification system, transporting them to a valve manifold in the valve cubicle of the sample preparation room. Thence the desired samples may be routed to any of three sample trains. The samples are obtained from the inlet and outlet sides of each demineralizer. The sample preparation room is immediately south of the east boiler chamber enclosure in the reactor service building.

Operational and analytical instruments required for chemical, physical, and radiochemical analyses of the reactor coolant are in the three identical sample trains, each of which is in a separately ventilated steel enclosure. The enclosures prevent reactor coolant from escaping into the sample preparation room if the sample train piping leaks.

Each sample train contains the following analytical instruments in addition to a sample vessel which is used to obtain high pressure samples of the reactor coolant:

Instrument	*Function*
Crud probe	Determines activity of insoluble corrosion and fission products in sample and accumulates these products for analysis.
Gross soluble activity monitor	Determines activity of soluble products in sample after insolubles have been removed.
Conductivity cell	Determines ionized soluble impurity concentration of sample.
pH cell	Determines hydrogen ion concentration of sample.
Pilot ion exchanger	Demineralizes coolant water as required for oxygen analyzer, and accumulates soluble contaminants in sample for analysis.
Oxygen analyzer	Determines quantity of oxygen dissolved in sample.

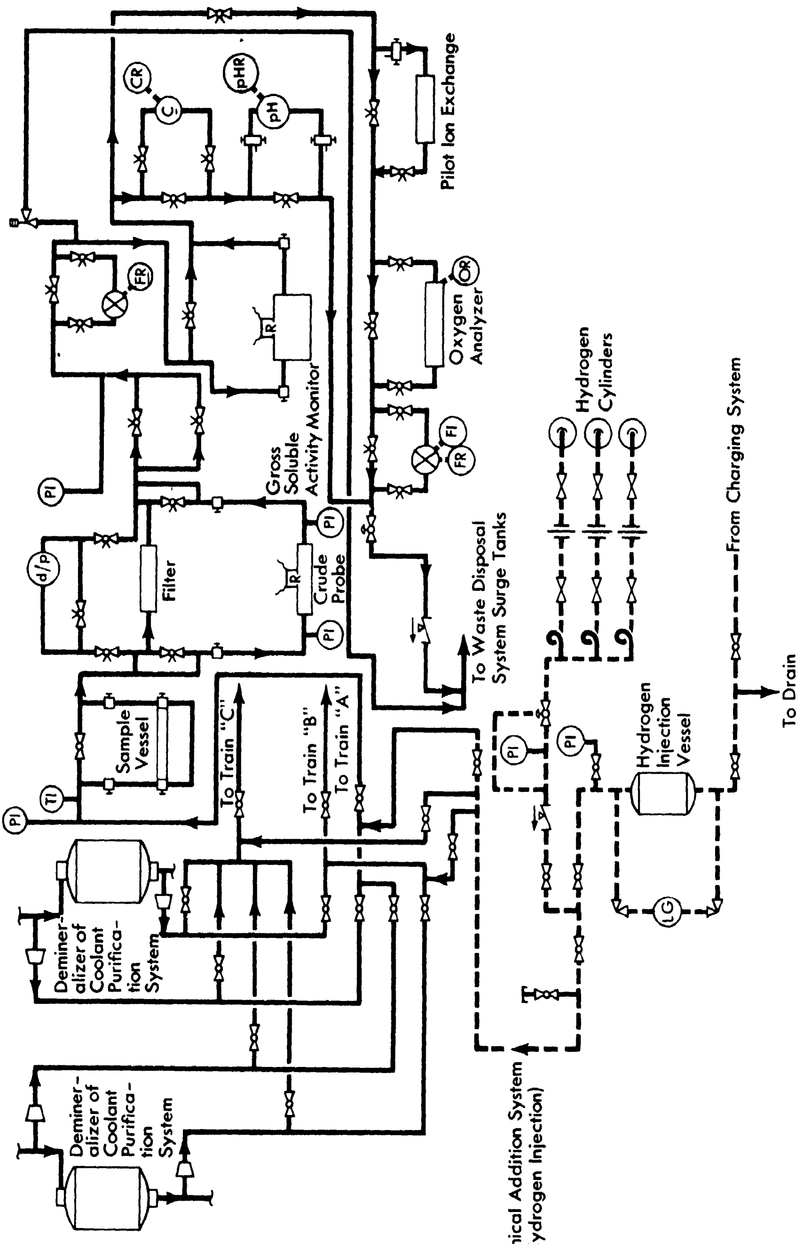

FIG. 8–11. Reactor coolant sampling system flow diagram. A portion of the coolant chemical addition system, discussed earlier in this chapter, is shown by dashed lines.

In addition to the components described above, each sample train contains a filter for use only when the crud probe (which also acts as a filter) is out of service. All corrosion and fission product particulate matter must be removed before valid information can be obtained from the conductivity and pH instruments downstream of the filter and crud probe. The trains contain operational instruments, valves, and piping required to control the sample flow path and flow rate.

After the samplings, the effluent from the system is discharged to the surge and decay tanks in the radioactive waste disposal system.

Design requirements. The system can withdraw samples of coolant from each of the coolant purification system loops for either continuous or intermittent monitoring in the sample trains. In addition to providing facilities for obtaining basic chemical, physical, and radiochemical data concerning the reactor coolant, the system provides permanent records of pH, conductivity, dissolved oxygen, soluble activity, and insoluble activity.

The components of the system upstream of the sample train throttle valves in the high pressure portion of the system have a design pressure of 2500 psig and a design temperature of 250°F. These conditions are compatible with those of the low temperature portion of the coolant purification system, to which they are attached. The remaining components in the low pressure portion of the sample train require design conditions of 200 psig and 250°F.

The system provides means for controlling the sample flow rate to each train as required for operational purposes to: (1) provide for adequate decay of N^{16} before the sample enters the trains and (2) attain turbulent flow conditions in the sample lines to provide representative samples of crud.

8-3.2 Failed element detection and location system. *Function.* The failed element detection and location system (Fig. 8-12) supplies the plant operator with information required to locate a failed fuel element of a natural uranium assembly within the reactor.

General description. Well-mixed samples from the effluent of each of the 113 UO_2 blanket assemblies are individually and simultaneously brought to a multiport valve. The multiport valve, at the top of the reactor vessel, selects two of these samples for passage through the monitoring channels; the remaining 111 samples are routed to a bypass line.

The two selected samples are routed individually through separate, but similar, sampling trains. Each sample flows through an excess flow check valve, a manual capped isolation valve, and one or two time delay coils or a bypass line selected to provide the proper sample transit time. The flow then passes through a monitoring station which consists of a holdup coil and the necessary instrumentation to detect delayed neutrons which

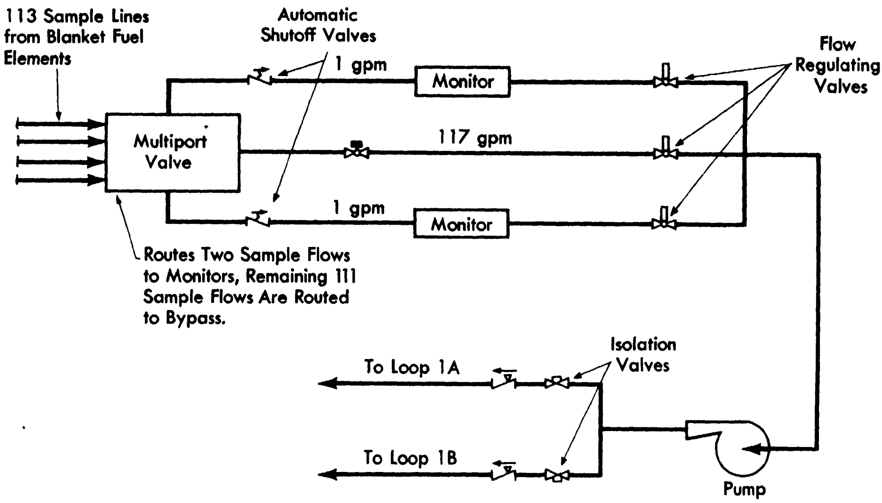

FIG. 8–12. Failed element detection and location system flow diagram.

are indicative of fuel element failure(s). After passing through a flow sensing element and a manual capped isolation valve, the two samples join with the bypass flow.

After leaving the multiport valve, the flow from the 111 samples not being monitored is routed through a hydraulically operated isolation valve, a flow sensing element, and a capped manual isolation valve to a junction with the monitored samples. The combined flow enters the suction of the system's canned motor circulating pump and is discharged into the reactor coolant system via loops 1A and 1B.

Means are provided for utilizing the sample trains interchangeably. A crossover line with a manual capped isolation valve connects sample trains 1 and 2 upstream of the time delay coils.

The various mechanical components are in the pressurizer area of the auxiliary chamber and in the reactor chamber. The delayed neutron activity monitors are outside the pressurizer shielding in the auxiliary chamber. The detectors, consisting of groups of BF_3 proportional counters, furnish signals to the computer circuitry in the system panel in the main control room.

Visual and audible alarms are given when the sample activity is between the normal operating level and a predetermined minimum undesirable level. Alarm contacts show when the sample activity rises above the minimum undesirable level. Panel instruments show which assembly is providing the alarm and which assemblies are currently being monitored.

After the two selected samples have been monitored for a predetermined and controllable time interval, the multiport valve is used to select another

pair of samples for monitoring. This can be done automatically until samples from all 113 assemblies have been monitored. The multiport valve controls will then cause the valve to repeat the sampling sequence automatically. This control mode is termed *automatic.* If desired, the multiport valve can be operated to select samples from a pair of assemblies out of sequence and for an indefinite period of time, instead of being used for the automatic sampling sequence just described. This is known as *manual* control.

The sample flow rates are recorded as the samples are being monitored. The flow recorders are also on the panel; visual and audible alarms are given if the flow rates are not as desired. In addition, local flow indication is provided near the flow regulating valves to adjust them to obtain the proper sample flow rates.

The bypass flow can be determined by the sample train No. 2 flow indicator, located near the flow regulating valve for the bypass system flow.

Design requirements. The system must obtain well-mixed samples of the effluent of two of the 113 blanket assemblies and transport them out of the pressure vessel to external delayed neutron monitors. The samples from the 111 assemblies not being carried to the monitors must be kept in continuous flow through suitable bypass lines and returned to the reactor coolant system. Also, operation of this system must not interfere with normal plant operation. The system must be capable of operating automatically and continuously for all 113 assemblies in a sequence of pairs with a 10-sec maximum switchover time from one pair to the adjacent pair.

Manual override of the automatic operation must be available at all times; that is, it must be possible at all times during system operation for the plant operator to select any one of the 57 pairs of assemblies for sampling and for any desired duration of sampling time. The maximum allowable switchover time from any pair to any other pair is three minutes during manual operation. Furthermore, there must be highly reliable assurance that the assembly indicated is the one actually being sampled.

The system design is based on an average requirement of a 1 gpm sample flow rate at reactor conditions of 1985 psi and 545°F for the 113 flow paths. The over-all leakage of coolant loop water into any assembly sample, while sampling that assembly, must not exceed 10% of the sample flow during its travel through the entire system.

The design of the external portion of the system is based upon a normal sample temperature of 540 to 560°F at the multiport valve, and upon an emergency temperature of 636°F, which may occur if the sampling or bypass flow is seriously reduced. Further, the external portion of the system must provide for a satisfactory restart of the system following such an emergency.

Indication of the sample flow is provided at the controlling station, and

recording of the flow is provided in the control room. The recorder chart identifies the assembly being sampled. An alarm is provided in case of any assembly sample flow rate below 0.9 gpm or above 1.1 gpm.

The delayed neutron monitors are designed to yield an optimum combination of efficiency and signal-to-background ratio.

The transit time for the sample with the longest flow path, from the exit of the blanket assembly to the entrance of the monitor channels, is adjustable to a maximum of 50 sec for the minimum flow rate. Alternative transit times of 65 sec and 80 sec, at the same pressure drop and flow rate, are also provided.

The monitoring units, flow regulating valves, and local flow indicators are accessible during reactor operation, but are designed for maximum reliability, so that only an occasional periodic inspection is required. No system routine operating function which requires entry to the plant container is permitted. All such routine functions and monitoring for failed elements must be performed outside the plant container.

It was necessary that the system not be influenced by the presence of corrosion inhibitor chemicals, pH control chemicals, or other additives intentionally inserted into the reactor coolant, and that the system not influence the action of these additives. Excepted from this requirement is the use of chemical shutdown additives which might interfere with the use of the delayed neutron monitors.

Also, it was necessary to design the system such that (a) it would not prevent or seriously interfere with the normal removal or replacement of any components within the reactor vessel, and (b) such removal or replacement would not jeopardize the equipment of the system.

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CHAPTER 9

CONTROL AND INSTRUMENTATION SYSTEMS

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CONTROL AND INSTRUMENTATION SYSTEMS*

9–1. Introduction

This chapter describes the control and instrumentation of the reactor plant. Control and instrumentation of the turbogenerator plant follow conventional central station practice and are covered only to the extent that they relate to or affect the reactor and its plant systems.

9–1.1 Functions of the PWR control and instrumentation systems. The functions of the control and instrumentation systems for the PWR reactor plant may be classified under six general headings:

(1) The control systems provide for automatic or manual *regulation of plant parameters,* such as reactor coolant temperature and pressure and reactor power level (reactivity adjustment), and *control of reactor plant equipment,* such as the coolant pumps and valves.

(2) The control systems provide *safety arrangements* which protect the reactor core from damage caused by improper operating conditions.

(3) The instrumentation systems provide *information to the operating personnel* to enable them to make decisions necessary to operate the reactor and its associated plant systems.

(4) The instrumentation systems provide *information to the automatic control systems* of the reactor plant for automatic regulation of plant parameters, and *information to the protection systems* to ensure that the plant is safe from damage without unnecessary interruptions to power output.

(5) The instrumentation systems may provide *information of a design support nature.* Such information may not be required to operate the plant, but may lead to further knowledge about the behavior of the plant and its components.

(6) The control and instrumentation systems *monitor the plant and surrounding area for radiation hazards* and, in some cases, provide automatic action to eliminate the possibility of radioactive contamination of the area.

The over-all function of the control and instrumentation systems is to regulate and monitor the reactor plant in such a way that it may

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* By N. E. Wilson, Westinghouse Bettis Plant, and J. C. Grigg, U. S. Atomic Energy Commission.

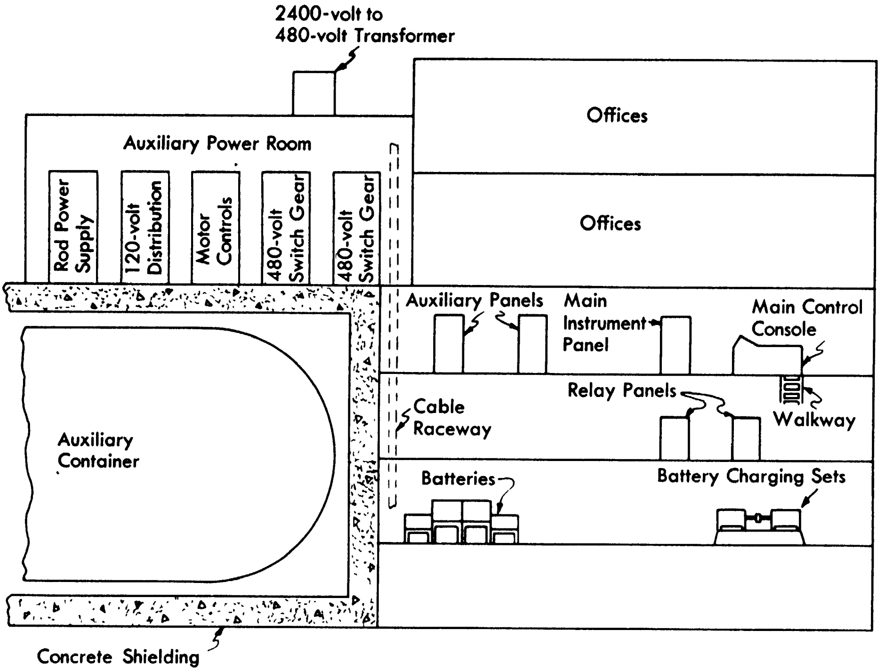


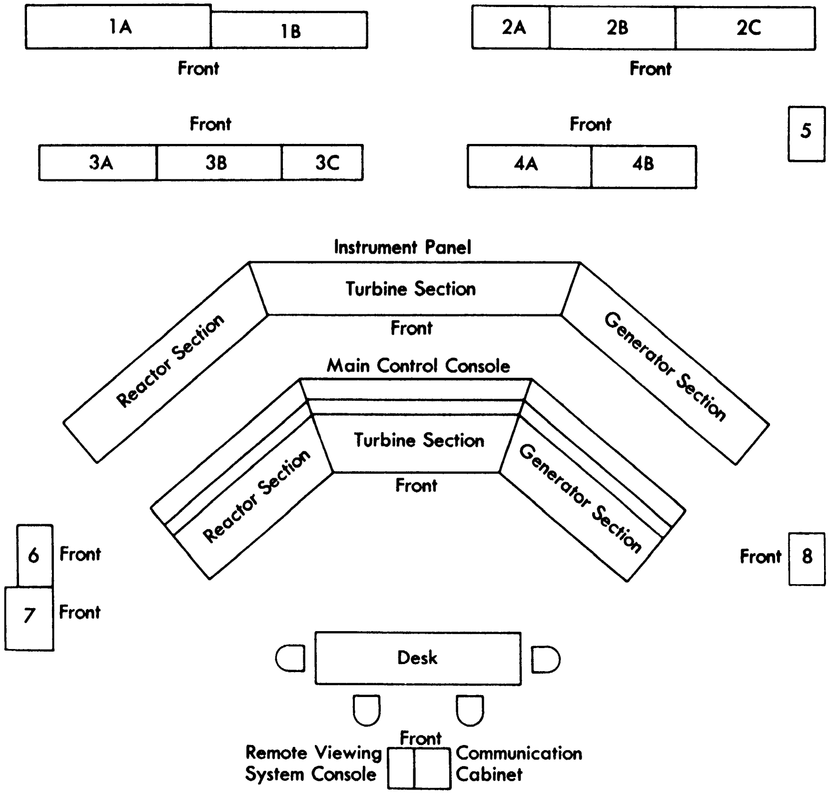
FIG. 9-1. Section through the station, showing location of control equipment.

serve its function as the heat source for a central station facility producing electric energy.

9-1.2 Control equipment arrangement. The sensing devices for the instrumentation systems and the devices regulated by the control systems are located throughout the reactor and turbogenerator plant areas. To permit close supervision of the entire station, the read-out equipment for the instruments and that for remote manual or automatic control are centrally located in the six-story turbogenerator service building between the east boiler chamber and the turbine building.

As shown in Fig. 9-1, control and instrumentation equipment occupies space on three levels of the building: battery equipment for the station vital control on the second level, automatic protective relaying equipment on the third, and the main operating control center on the fourth level. Administrative and supervisory offices occupy the fifth and sixth levels. Control equipment arrangement in the main operating control center is shown in Fig. 9-2.

Equipment for power distribution throughout the reactor portion of the station is immediately adjacent to the turbogenerator service building, one level above the main control room. Power distribution circuit breakers



Legend

1A Auxiliary Rod Control Panel
1B Nuclear Instrumentation Panel
2A Reactor Power and Temperature Control Panel
2B Auxiliary Instrument Panel
2C Auxiliary Control Board
3A Operational Radiation Monitoring System
3B Core Instrumentation System
3C Failed Element Detection and Location System
4A Reactor 480-volt Substation Control Panel
4B Turbine Room and Screen House 480-volt Substation Control Panel
5 Pump Noise Monitoring Cabinet
6 Battery Charger Control Panel
7 Boiler Water Level TV Monitor
8 138-kv Instrument Panel

FIG. 9-2. Plan view of the main operating control center, showing control equipment arrangement.

in the equipment in the auxiliary power room are remotely controlled from panels located in the main control center.

9-1.3 Control power supply. Power to operate all electrical auxiliaries in the reactor plant comes from four 500-kva, 2400 to 480-volt, outdoor station service transformers on the roof of the auxiliary power room as shown in Fig. 9-1. The connection of these transformers into the main 2400-volt station service system is shown in Fig. 9-3.

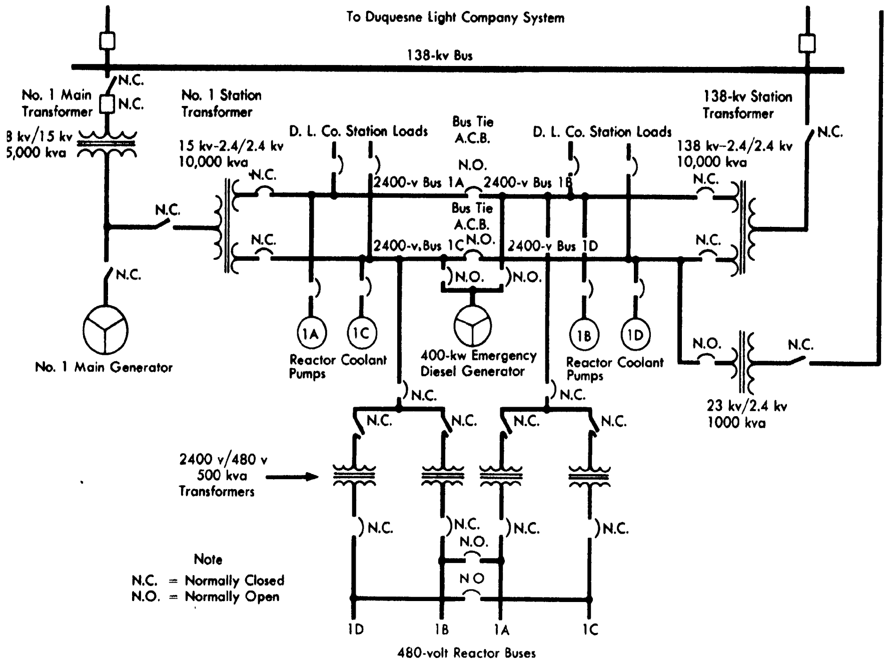


FIG. 9–3. Shippingport station service power connections.

The important feature of this system, from the standpoint of reactor safety, is that two sources of station service power are supplied. One, the 138-kv transmission line bus, has two transmission line connections into the power system. The second source is the main station generator. Vital auxiliaries such as coolant pumps, charging pumps, rod control motor-generator sets, etc., are duplicated and divided between the two sources. Bus ties between pairs of 2400 and 480-volt station service buses are open during power operation.

9–1.4 Scope of the control and instrumentation systems. Over-all aspects of the PWR control and instrumentation are discussed in this chapter under the following classifications:

(1) *Reactor instrumentation* includes instruments providing continuous control signals and information to the operator on the status of the nuclear reaction, both as to reactor power and rate of change of reactor power. It also includes means for determining the distribution of power production and of temperatures within the core.

(2) *Reactor plant instrumentation* principally includes sensing elements and their read-out equipment for coolant pressure, coolant flow, coolant temperature, and pressurizer level. It also includes measurement of

pressures, temperatures, flows, water levels, etc., of the auxiliary systems discussed in Chapter 8.

(3) The *reactor control systems* include the means for establishing the nuclear reaction at a sustained rate, for increasing the rate to the level required by the plant and maintaining it there, for adjusting reactor power to meet plant or safety demands, and for adjusting reactivity to offset the internal reactivity changes that occur because of changing plant demands.

(4) The *reactor plant control systems* include the circuitry which regulates coolant flow and pressure. The systems also control valves and pumps in the auxiliary systems.

(5) The *reactor protection system* includes the means not only of providing protection but also of analyzing the reactor's behavior under transient conditions. Such analysis is necessary to select the parameters by which dangerous conditions are detected and to determine rod insertion signal points.

(6) The *reactor plant monitoring systems* detect (a) excessive radioactivity levels in various areas and in various plant effluents, such as air from the plant container, water from the waste disposal system, etc.; (b) radioactivity level of the reactor coolant, indication of high level inferring a leak not otherwise detected; and (c) various temperatures in reactor plant components, giving an indication of component failure. Closed circuit television is also provided for viewing important areas of the container interior during plant operation.

9-2. Description of the Control and Instrumentation Systems

9-2.1 Reactor instrumentation. The most vital of the reactor instrumentation systems is that which monitors neutron flux level of the core. Since the average neutron flux level of the entire core in neutrons/cm^2·sec is a direct measure of the heat output of the core, it becomes important to measure this quantity and its derivative or rate of change. It is not possible to measure the average flux absolutely, since it is a quantity made up of the neutron densities of every unit volume of the core. Instead, a representative sample must be measured.

In the Shippingport reactor it was not considered feasible to insert neutron flux detectors into the core itself because:

(1) The *range of the quantity to be measured* from shutdown to full power varies over a ratio of approximately 1 to 10^9 and a detector suitable for measuring the lowest neutron flux encountered (shutdown and initial critical flux) would not be able to withstand the flux at full power without damage. Hence, a detector would have to be withdrawn as the reactor approached full power.

(2) *Providing wells in the vessel* designed for 2000 psi or, alternatively,

providing a sealed well with a sealed retracting mechanism and coaxial electrical leads through the seal for the detector, is a formidable problem.

Thus the neutron detectors for the reactor are outside the reactor pressure vessel, approximately parallel to the vertical axis of the core in the body of water forming the neutron shield. They therefore measure the fraction of neutrons that leak through the neutron reflector and thermal shields and through the pressure vessel wall. This fraction is approximately 1/10,000 the average neutron flux density in the core.

Although problems were avoided by locating the neutron flux detectors outside the reactor pressure vessel, others were thereby encountered. They are:

(1) The *small fraction of reactor shutdown flux* reaching the detector is on the order of 1 to 20 neutrons/cm^2·sec. The number depends on the temperature of the reactor coolant and the age of the neutron sources. To provide meaningful information, this small flux requires sensitive detectors and circuitry with low electrical noise characteristics.

(2) It is *difficult to predict the value of detector flux* in advance of reactor operation because of the complex nature of the attenuation through the core blanket assemblies, reflector, thermal shield, vessel wall, and neutron shield tank.

The problem of low detector flux can be overcome only by utilizing highly sensitive neutron detectors and carefully shielding cables and equipment from external electrical noise. A value of 1 neutron/cm^2·sec is about the lowest value that can give significant information above normal background noise and cosmic radiation.

Calculations were made to provide prior knowledge of the attenuation and hence of detector flux; to confirm these calculations, measurements were made on critical experiments that had a partial mockup of the vessel material included. Even with such experiments, a large area of uncertainty remained. After operation began the detectors were modified to reduce sensitivity by "shadowing" them with material containing boron.

Technically, it may not be essential to measure the lowest levels of neutron flux. That is, the flux from the neutron sources and the core flux at very low levels of operation need not be "seen" by the instruments, since power under these conditions is on the order of a few watts to a few kilowatts. Even so, it is considered that, for safe operation, the operator should have full knowledge of the behavior of the reactor from the first neutron multiplication as rods are withdrawn. Therefore, the entire range of core operation is monitored. To cover values of a quantity ranging from 10^4 to 10^{13} required the application of two ranges of instruments. For the first, a boron trifluoride detector covers the reactor operation from its shutdown condition (source strength) of approximately 10^4 neutrons/cm^2·sec to 10^8 neutrons/cm^2·sec. A gamma compensated ion chamber

covers the range from 10^7 to 10^{14} neutrons/cm^2·sec. (These are average reactor flux levels, not detector levels.) Thus the instrument ranges overlap so that the reactor neutron flux conditions are indicated to the operator at all times. These detectors are described more fully in Chapter 12.

The instrumentation computes the *rate of change of flux* and presents it as decades per minute* of flux increase or decrease. It is thus inversely proportional to the familiar "period" (the time in seconds required for the flux to increase by a factor of *e*) and is numerically equal to it for a period of 5 sec, which is a rate of change of five decades per minute. For PWR, a rate of 1.74 decades per minute was selected as the limiting rate to provide maximum protection during reactor startup without requiring excessive time to bring the plant to power. At faster rates of power increase, the rods are automatically inserted. A rate-of-change signal is also used as a stabilizing element in the reactor power and temperature regulating system.

As noted above, nuclear instruments measure a representative sample of the core neutron flux. Variations in positions of the control rods and in fission product poisoning with power level changes cause changes in the ratio of the flux at the detectors to the core average flux. For this reason, and because it is desired not to have the operator dependent on only one instrument, there are four independent channels of neutron flux instrumentation, placed around the core so that each channel views a different segment of the core. When any two of these four channels indicate excessive neutron flux levels, the safety circuits cause automatic shutdown of the reactor. These circuits are described more completely in the discussion of the reactor protection system.

To provide information on the thermal performance of the core, to verify design data, and to evaluate performance as a function of core age, considerable instrumentation was installed to indicate and record representative core temperature and flow data.

Sixteen seed and 20 blanket fuel assemblies are instrumented with differential pressure cells across the fuel assembly orifice plates to measure flow. Sixteen seed and 27 blanket fuel assemblies have thermocouples placed to measure the temperature of the water leaving the fuel assembly. Eleven thermocouples distributed about the bottom of the core give representative core inlet water temperatures. These thermocouples (exit and inlet) furnish temperature rise data for use with the flow data to compute power distribution in various regions of the core. The arrangement of the differential pressure cells for flow instrumentation is shown in Fig. 9-4. The instrument piping to the cells is also shown in this illus-

* Decades per minute refers to the rate of change of the exponent of ten in the number expressing the neutron flux level.

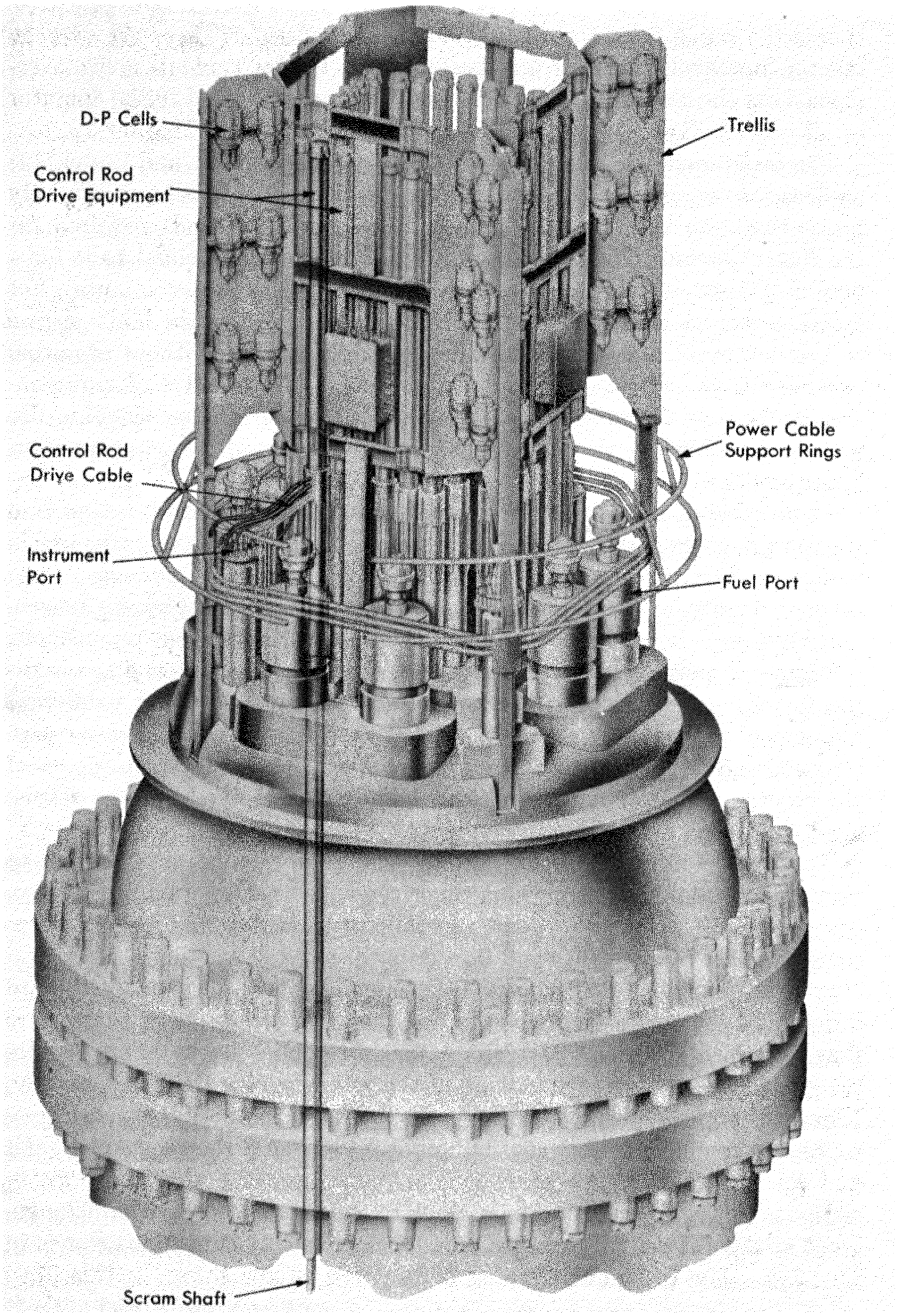


FIG. 9–4. Arrangement of core instrument and control rod equipment on the reactor vessel head.

tration. All flow and temperature sensing elements are connected to multipoint recorders in the main control center. The panel for these recorders, which are a part of the core instrumentation system, is shown in Fig. 9-2.

In addition to measurements of flow and temperature, the core is also instrumented for the detection of failed blanket fuel assemblies. Water from the 113 blanket assemblies is sampled in sequence by the automatic operation of a multiport valve. (The system for control of this valve is described in detail in Chapter 8.) The sample of water is passed through a coil which is surrounded by 44 boron trifluoride proportional counters like those used in the source range for nuclear instrumentation. The detectors monitor the sample for the presence of the delayed neutron emitters Br^{87} and I^{137}, having half-lives of 56 and 22 sec, respectively. The 44 counters provide signals through pulse integrators and amplifiers to recorders on the failed element detection and location panel in the main control center (see Fig. 9-2).

9-2.2 Reactor plant instrumentation. In a pressurized water reactor plant the vital instrumentation, other than that of the reactor itself, is concerned with those quantities relating directly to the reactor coolant: *reactor coolant temperature, pressure,* and *flow.* Directly related to these quantities is the *coolant level in the pressurizer.* The locations of detectors for these measurements are shown in Fig. 2-1. The signals obtained from these detectors are displayed on indicating instruments grouped for the operator's convenience on the instrument section of the main control console of the reactor plant (Fig. 9-5).

In each of the four reactor coolant loops, one resistance thermometer element is near the coolant outlet from the reactor vessel, and another is located near the outlet from the steam generator. Thus these two elements of each loop sense the coolant hot leg temperature (T_h) and cold leg temperature (T_c) at points as close as possible to those devices in the system which create changes in these temperatures. For each coolant loop, there is a resistance thermometer element located in the steam generator secondary water. Temperatures obtained are not read out directly, but used in the temperature interlock device to prevent starting the loop pump if loop temperature is substantially lower than that of the operating loops.

Each coolant loop is instrumented for flow by units which measure the pressure drop across a calibrated venturi, and for pressure by static pressure units. Reactor coolant pressure is measured in the pressurizer. A pressure instrument is also connected into the coolant piping on the reactor vessel, ahead of the shutoff valves. The read-out portion of this instrumentation is located on the control console (see Fig. 9-5). This

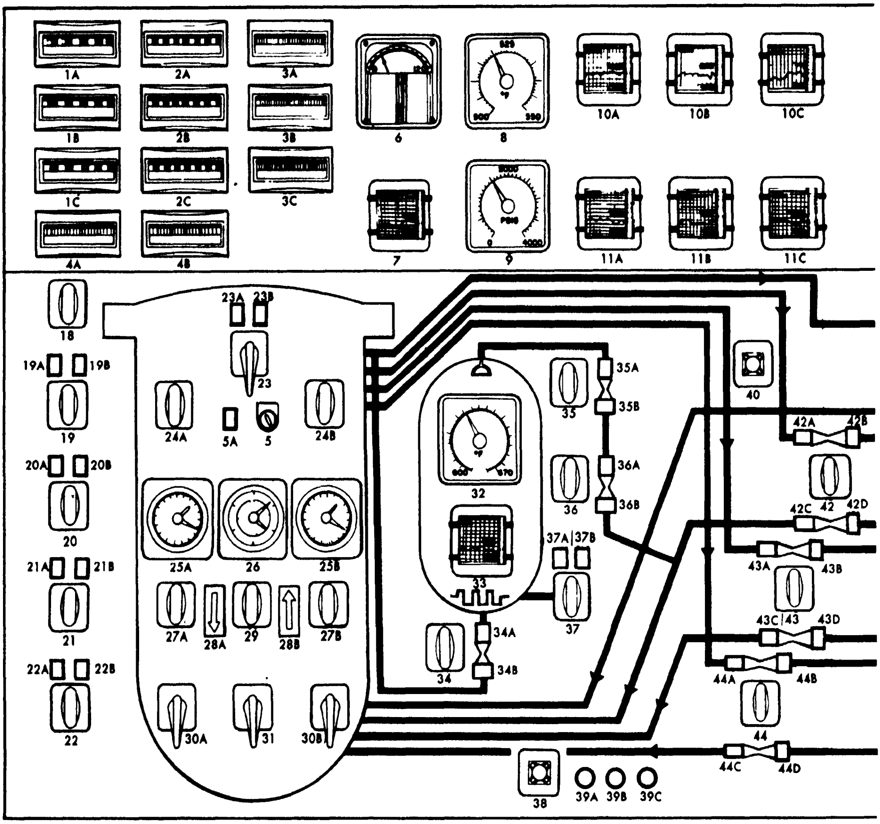


FIG. 9-5. Actual arrangement of instruments and controls on reactor section of main control console.

LEGEND

1A LOG LEVEL—SOURCE RANGE—A CHANNEL
1B LOG LEVEL—SOURCE RANGE—B CHANNEL
1C LOG LEVEL—SOURCE RANGE—C CHANNEL
2A LOG LEVEL—INTERMEDIATE RANGE—A CHANNEL
2B LOG LEVEL—INTERMEDIATE RANGE—B CHANNEL
2C LOG LEVEL—INTERMEDIATE RANGE—C CHANNEL
3A LINEAR LEVEL—POWER RANGE—A CHANNEL
3B LINEAR LEVEL—POWER RANGE—B CHANNEL
3C LINEAR LEVEL—POWER RANGE—C CHANNEL
4A START-UP RATE—SOURCE RANGE
4B START-UP RATE—INTERMEDIATE RANGE
5 TEST PERMISSIVE
5A REACTOR PROTECTION SYSTEM IN TEST (RED)
6 NO. 1 MAIN GENERATOR
7 PRESSURIZER PRESSURE (1800 TO 2300 PSIG)
8 AVERAGE COOLANT TEMP
9 REACTOR PRESSURE
10A 1A COOLANT LOOP T_H (475 TO 575°F)
10B 1B COOLANT LOOP T_H (475 TO 575°F)
10C 1C COOLANT LOOP T_H (475 TO 575°F)
10D 1D COOLANT LOOP T_H (475 TO 575°F)
11A 1A COOLANT LOOP T_C (475 TO 575°F)
11B 1B COOLANT LOOP T_C (475 TO 575°F)
11C 1C COOLANT LOOP T_C (475 TO 575°F)
11D 1D COOLANT LOOP T_C (475 TO 575°F)
12A 1A COOLANT LOOP FLOW
12B 1B COOLANT LOOP FLOW
12C 1C COOLANT LOOP FLOW
12D 1D COOLANT LOOP FLOW
13A 1A COOLANT LOOP PRESSURE
13B 1B COOLANT LOOP PRESSURE
13C 1C COOLANT LOOP PRESSURE
13D 1D COOLANT LOOP PRESSURE
14A 1A CHARGING PUMP DISCHARGE
14B 1B CHARGING PUMP DISCHARGE
15 FLASH TANK LEVEL
16 FLASH TANK TEMP
17 FLASH TANK OPERATION
17A FLASH TANK SPRAY VALVE OPEN (RED)
17B FLASH TANK SPRAY VALVE CLOSED (GREEN)
17C FLASH TANK SPRAY PUMP OFF (GREEN)
17D FLASH TANK SPRAY PUMP ON (RED)
17E FLASH TANK INLET VALVE OPEN (RED)
17F FLASH TANK INLET VALVE CLOSED (GREEN)
18 SAFETY INJECTION
19 NO. 1 REACT RELIEF VALVES TRANSFER
19A NO. 1 REACT RELIEF VALVES TRANSFER A OPEN (RED)
19B NO. 1 REACT RELIEF VALVES TRANSFER B OPEN (RED)
20 NO. 2 REACT RELIEF VALVES TRANSFER
20A NO. 2 REACT RELIEF VALVES TRANSFER A OPEN (RED)
20B NO. 2 REACT RELIEF VALVES TRANSFER B OPEN (RED)
21 PRESSURIZER RELIEF VALVES TRANSFER
21A PRESSURIZER RELIEF VALVES TRANSFER A OPEN (RED)
21B PRESSURIZER RELIEF VALVES TRANSFER B OPEN (RED)
22 PRESSURIZER PILOT RELIEF STOP VALVE
22A PRESSURIZER PILOT RELIEF STOP VALVE OPEN (GREEN)
22B PRESSURIZER PILOT RELIEF STOP VALVE CLOSED (RED)
23 SAFETY SHUTDOWN
23A SAFETY SHUTDOWN TRIP (GREEN)
23B SAFETY SHUTDOWN CLOSED (RED)
24A SOURCE RANGE
24B INTERMEDIATE RANGE
25A ROD POSITION NO. 1 SPARE BUS
25B ROD POSITION NO. 2 SPARE BUS
26 ROD MOTION TOTALIZER
27A NO. 1 SELECTOR INVERTER OPERATION
27B NO. 2 SELECTOR INVERTER OPERATION
28A RODS IN (BLUE ARROW)
28B RODS OUT (BLUE ARROW)
29 ROD OPERATION SELECTOR
30A NO. 1 CONTROL INVERTER OPERATION
30B NO. 2 CONTROL INVERTER OPERATION
31 MANUAL ROD CONTROL
32 PRESSURIZER TEMPERATURE
33 PRESSURIZER LEVEL (25" TO 125")
34 SURGE ISOLATION
34A SURGE ISOLATION OPEN (RED)
34B SURGE ISOLATION CLOSED (GREEN)
35 SPRAY CONTROL

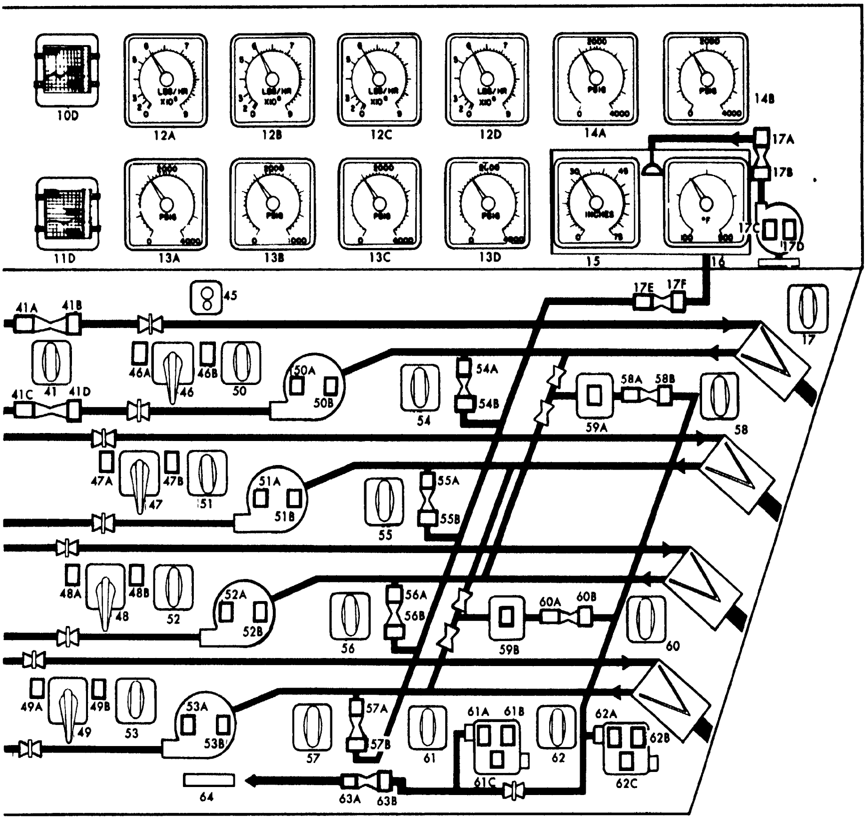


FIGURE 9–5. *(Continued)*

LEGEND

35A SPRAY CONTROL OPEN (RED)
35B SPRAY CONTROL CLOSED (GREEN)
36 SPRAY ISOLATION
36A SPRAY ISOLATION OPEN (RED)
36B SPRAY ISOLATION CLOSED (GREEN)
37 PRESSURIZER HEATERS
37A PRESSURIZER HEATERS OFF (GREEN)
37B PRESSURIZER HEATERS ON (RED)
38 STAND-BY ALARM
39A TEST
39B RESET
39C ACKNOWLEDGE
40 TEMPERATURE INTERLOCK
41 1A LOOP HYD VALVES
41A 1A LOOP INLET VALVE OPEN (RED)
41B 1A LOOP INLET VALVE CLOSED (GREEN)
41C 1A LOOP OUTLET VALVE OPEN (RED)
41D 1A LOOP OUTLET VALVE CLOSED (GREEN)
42 1B LOOP HYD VALVES
42A 1B LOOP INLET VALVE OPEN (RED)
42B 1B LOOP INLET VALVE CLOSED (GREEN)
42C 1B LOOP OUTLET VALVE OPEN (RED)
42D 1B LOOP OUTLET VALVE CLOSED (GREEN)
43 1C LOOP HYD VALVES
43A 1C LOOP INLET VALVE OPEN (RED)
43B 1C LOOP INLET VALVE CLOSED (GREEN)
43C 1C LOOP OUTLET VALVE OPEN (RED)
43D 1C LOOP OUTLET VALVE CLOSED (GREEN)
44 1D LOOP HYD VALVES
44A 1D LOOP INLET VALVE OPEN (RED)
44B 1D LOOP INLET VALVE CLOSED (GREEN)
44C 1D LOOP OUTLET VALVE OPEN (RED)
44D 1D LOOP OUTLET VALVE CLOSED (GREEN)
45 AIR LOADING VALVE
46 1A COOLANT PUMP
46A 1A COOLANT PUMP OFF (GREEN)
46B 1A COOLANT PUMP ON (RED)
47 1B COOLANT PUMP
47A 1B COOLANT PUMP OFF (GREEN)
47B 1B COOLANT PUMP ON (RED)
48 1C COOLANT PUMP
48A 1C COOLANT PUMP OFF (GREEN)
48B 1C COOLANT PUMP ON (RED)
49 1D COOLANT PUMP
49A 1D COOLANT PUMP OFF (GREEN)
49B 1D COOLANT PUMP ON (RED)
50 1A COOLANT PUMP SPEED
50A 1A COOLANT PUMP SPEED SLOW (RED)
50B 1A COOLANT PUMP SPEED FAST (RED)
51 1B COOLANT PUMP SPEED
51A 1B COOLANT PUMP SPEED SLOW (RED)
51B 1B COOLANT PUMP SPEED FAST (RED)
52 1C COOLANT PUMP SPEED
52A 1C COOLANT PUMP SPEED SLOW (RED)
52B 1C COOLANT PUMP SPEED FAST (RED)
53 1D COOLANT PUMP SPEED
53A 1D COOLANT PUMP SPEED SLOW (RED)
53B 1D COOLANT PUMP SPEED FAST (RED)
54 1A COOLANT LOOP DRAIN
54A 1A COOLANT LOOP DRAIN OPEN (RED)
54B 1A COOLANT LOOP DRAIN CLOSED (GREEN)
55 1B COOLANT LOOP DRAIN
55A 1B COOLANT LOOP DRAIN OPEN (RED)
55B 1B COOLANT LOOP DRAIN CLOSED (GREEN)
56 1C COOLANT LOOP DRAIN
56A 1C COOLANT LOOP DRAIN OPEN (RED)
56B 1C COOLANT LOOP DRAIN CLOSED (GREEN)
57 1D COOLANT LOOP DRAIN
57A 1D COOLANT LOOP DRAIN OPEN (RED)
57B 1D COOLANT LOOP DRAIN CLOSED (GREEN)
58 1AC LOOPS CHARGING VALVE
58A 1AC LOOPS CHARGING VALVE OPEN (RED)
58B 1AC LOOPS CHARGING VALVE CLOSED (GREEN)
59A 1AC PURIFICATION LOOP FLOW
59B 1BD PURIFICATION LOOP FLOW
60 1BD LOOPS CHARGING VALVE
60A 1BD LOOPS CHARGING VALVE OPEN (RED)
60B 1BD LOOPS CHARGING VALVE CLOSED (GREEN)
61 1A CHARGING PUMP
61A 1A CHARGING PUMP SLOW (RED)
61B 1A CHARGING PUMP FAST (RED)
61C 1A CHARGING PUMP OFF (GREEN)
62 1B CHARGING PUMP
62A 1B CHARGING PUMP SLOW (RED)
62B 1B CHARGING PUMP FAST (RED)
62C 1B CHARGING PUMP OFF (GREEN)
63A WATER SUPPLY STOP VALVE OPEN (RED)
63B WATER SUPPLY STOP VALVE CLOSED (GREEN)
64 VALVE OPERATING SYSTEM (LEGEND)

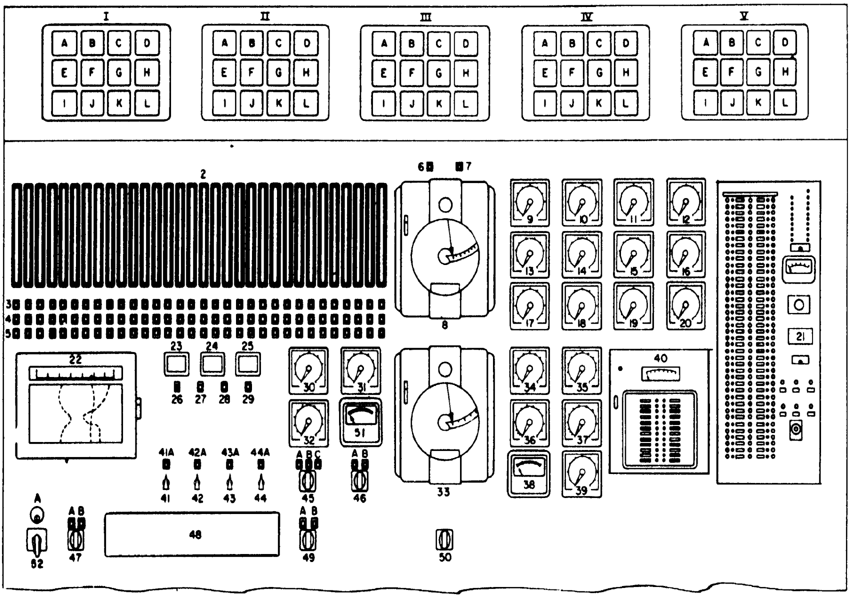


FIG. 9-6. Actual arrangement of instruments on reactor instrument panel.

LEGEND

ANNUNCIATOR WINDOWS—REACTOR SECTION

I GROUP I ANNUNCIATOR

- A SAFETY INSERTION OPERATION
- B SAFETY SHUTDOWN OPERATION
- C CORE INST TROUBLE
- D REACTOR COOLANT SYSTEM HIGH TEMP
- E SINGLE ROD DRIVE GENERATOR OPERATING
- F ROD DRIVE MG SET HIGH TEMP.
- G ROD CONTROL VOLTAGE LOSS
- H SPARE
- I STEAM GENERATOR LEAK
- J AIR PARTICLE DETECTOR MECH. FAILURE
- K PLANT CONTAINER AIR CONTAMINATION
- L REACTOR BELOW MINIMUM POWER

II GROUP II ANNUNCIATORS

- A REACTOR COOLANT SYSTEM LOW PRESSURE
- B PRESSURIZER LEVEL HIGH OR LOW
- C SAFETY INJECTION CHAMBER FLOODED
- D SPARE
- E REACTOR COOLANT SYSTEM HIGH PRESSURE
- F VALVE OPERATING AIR HEADER LOW PRESSURE
- G VALVE OPERATING FLASK LEVEL HIGH OR LOW
- H MAIN COOLANT VALVE DRIFT
- I RELIEF VALVE OPERATION
- J REACTOR PROTECTION SYSTEM TEST
- K COOLANT SAMPLING TROUBLE
- L NO LOW PRESSURE PROTECTION

III GROUP III ANNUNCIATORS

- A TEMP. MONITOR OPERATION
- B SPARE
- C SPARE
- D SPARE
- E COMPONENT COOLING PUMP TRIP
- F COMPONENT COOLING WATER EXP. TANK LEVEL
- G NEUTRON SHEILD EXP. TANK LOW LEVEL
- H COMPONENT COOLING WATER HIGH ACTIVITY
- I 1A LOOP LOW FLOW
- J 1B LOOP LOW FLOW
- K 1C LOOP LOW FLOW
- L 1D LOOP LOW FLOW

FIGURE 9-6. (*Continued*)

IV GROUP IV ANNUNCIATORS
- A VENTILATION SYSTEM VALVE FAILURE
- B BLOW-OFF TANK LEVEL HIGH OR LOW
- C FLASH TANK LEVEL HIGH OR LOW
- D WASTE DISPOSAL TROUBLE
- E VENTILATION SYSTEM TROUBLE
- F BLOW-OFF TANK PRESSURE
- G FLASH TANK PRESSURE
- H CANAL WATER STORAGE TANK LOW LEVEL
- I FLASH TANK HYDROGEN CONCENTRATION
- J CHARGING PUMP HIGH PRESSURE
- K CHARGING WATER HIGH CONDUCTIVITY
- L PRIMARY WATER STORAGE TANK LOW LEVEL

V GROUP V ANNUNCIATORS
- A 2400/480 VOLT 1A TRANSFORMER OVERLOAD
- B 2400/480 VOLT 1B TRANSFORMER OVERLOAD
- C 2400/480 VOLT 1C TRANSFORMER OVERLOAD
- D 2400/480 VOLT 1D TRANSFORMER OVERLOAD
- E 480 V AC B REACTOR BUS SUPPLY
- F 480 V AC B REACTOR BUS FEEDER
- G SPARE
- H SPARE
- I SPARE
- J SPARE
- K SPARE
- L SPARE

2 ROD POSITION INDICATORS (RED)
3 ROD BOTTOM INDICATORS (GREEN)
4 UPPER PROGRAMMING LIMIT INDICATORS (BLUE)
5 ROD ON SPARE BUS INDICATORS (AMBER)
6 SAFETY INJECTION 1A HEADER FLOW (RED)
7 SAFETY INJECTION 1B HEADER FLOW (RED)
8 PRESSURIZER PRESSURE (0-3000 PSIG)
9 1A COOLANT LOOP HEAT EXCHANGER D/P (0-50 PSID)
10 1B COOLANT LOOP HEAT EXCHANGER D/P (0-50 PSID)
11 1C COOLANT LOOP HEAT EXCHANGER D/P (0-50 PSID)
12 1D COOLANT LOOP HEAT EXCHANGER D/P (0-50 PSID)
13 1A COOLANT PUMP D/P (0-225 PSID)
14 1B COOLANT PUMP D/P (0-225 PSID)
15 1C COOLANT PUMP D/P (0-225 PSID)
16 1D COOLANT PUMP D/P (0-225 PSID)
17 1A COOLANT LOOP REACTOR D/P (0-100 PSID)
18 1B COOLANT LOOP REACTOR D/P (0-100 PSID)
19 1AC PURIFICATION LOOP FLOW (0-35000 LB/HR)
20 1BD PURIFICATION LOOP FLOW (0-35000 LB/HR)
21 TEMPERATURE MONITOR (80 POINTS 0-700°F)
22 TEMPERATURE MONITOR (COOLANT LOOP T_H'S T_C'S AND PRESSURIZER WATER 0-700°F)
23 NO. 1 MG SET ON (RED)
24 NO. 2 MG SET ON (RED)
25 NO. 3 MG SET ON (RED)
26 1B 480 V REACTOR BUS ENERGIZED (RED)
27 1A 480 V REACTOR BUS ENERGIZED)RED)
28 1D 480 V REACTOR BUS ENERGIZED (RED)
29 1C 480 V REACTOR BUS ENERGIZED (RED)
30 VALVE OPERATING WATER HEADER PRESSURE (0-4000 PSIG)
31 VALVE OPERATING AIR HEADER PRESSURE (2500-3000 PSIG)
32 VALVE OPERATING WATER FLASK LEVEL (0-300 INCHES)
33 PRESSURIZER LEVEL (0-200 INCHES)
34 FLASH TANK PRESSURE (0-600 PSIG)
35 BLOW-OFF TANK PRESSURE (0-600 PSIG)
36 FLASH TANK PRESSURE (0-3 IN. Hg VAC., 0-10 PSIG PRESS)
37 BLOW-OFF TANK PRESSURE (0-3 IN. Hg VAC., 0-10 PSIG PRESS)
38 RADIOACTIVE WASTE DISPOSAL HYDROGEN CONCENTRATION (0-4 %)
39 BLOW-OFF TANK LEVEL (0-100 INCHES)
40 TEMPERATURE INDICATOR (24 POINTS 0-700°F)
41 T_B INTERLOCK THERMOMETER 1A HEAT EXCHANGER
41A T_B INTERLOCK THERMOMETER 1A HEAT EXCHANGER CALIBRATE (GREEN)
42 T_B INTERLOCK THERMOMETER 1B HEAT EXCHANGER
42A T_B INTERLOCK THERMOMETER 1B HEAT EXCHANGER CALIBRATE (GREEN)
43 T_B INTERLOCK THERMOMETER 1C HEAT EXCHANGER
43A T_B INTERLOCK THERMOMETER 1C HEAT EXCHANGER CALIBRATE (GREEN)
44 T_B INTERLOCK THERMOMETER 1D HEAT EXCHANGER
44A T_B INTERLOCK THERMOMETER 1D HEAT EXCHANGER CALIBRATE (GREEN)
45 AIR LOADING VALVE OPERATION (20-0)
45A AIR LOADING VALVE OPERATION VENT (GREEN)
45B AIR LOADING VALVE OPERATION LOAD (RED)
45C AIR LOADING VALVE OPERATION BLOCK (BLUE)
46 VALVE OPERATING AIR COMPRESSOR (88-0)
46A VALVE OPERATING AIR COMPRESSOR OFF (GREEN)
46B VALVE OPERATING AIR COMPRESSOR ON (RED)
47 EMERGENCY POWER OPERATION
47A EMERGENCY POWER OPERATION BATTERY (RED)
47B EMERGENCY POWER OPERATION NORMAL (RED)
48 CALIBRATING AND BOILER SECONDARY THERMOMETER TEST TERMINALS
49 EMERGENCY AIR LOAD OPERATION (20-0B)
49A EMERGENCY AIR LOAD OPERATION NORMAL (GREEN)
49B EMERGENCY AIR LOAD OPERATION EMERGENCY (RED)
50 LOW PRESSURE PROTECTION (S-13)
51 INTERLOCK TEMPERATURE DIFFERENCE METER
52 TEMPERATURE INTERLOCK OVERRIDE SWITCH
52A TEMPERATURE INTERLOCK OVERRIDE SWITCH LOCK

instrumentation also has control and alarm functions which are discussed later.

Plant information not requiring immediate attention by the operator is presented on the main instrument panel directly behind the console. This is shown in Fig. 9–6.

9–2.3 Reactor control systems. *Reactivity control.* The neutron absorbing hafnium rods used to regulate the reactivity of the core are described in Chapter 4. The reactivity control system, consisting of the control rod drive motors, their power supplies and control circuits, was developed to meet the safety requirements described in Chapter 11.

To meet these requirements special motors are used to move the rods out of and into the core at slow rates for regulation, and to release them for rapid insertion for core protective action. To control the rods and thereby regulate the operation of the reactor, electrical power of unique characteristics is applied to these rod drive motors.

The motor is special for the following reasons:

(1) Because the reactor coolant system must be completely sealed, *no packed gland* for the rod extension is permitted. Thus the rotating part of the motor and its means of translating its rotary motion to linear motion of the rod must be inside the pressure barrier of the reactor coolant system. The motor is therefore canned and operates in the water of the coolant system, which lubricates moving parts.

(2) Reducing gears operating in the reactor coolant are not considered practical, so the *simplest kinetic coupling* must be used. The mechanism consists of a roller nut directly attached to the rotor and operating directly on a lead screw extension of the rod.

(3) The direct coupling, combined with the slow speed requirement for the rod, demands a *very slow speed motor.* Normal speed is 22 rpm.

(4) Because the rotor is inside the reactor coolant pressure barrier, *no electrical power* may be applied to the rotor. Hence, torque is applied by the action of magnetic flux established solely by windings on the stator. This flux operates through the can between stator and rotor which forms the hermetic seal for the reactor coolant.

(5) The control rod lead screw *must be able to disengage quickly* from the roller nut to drop the rod into the core. To accomplish this, the roller nut is split in halves. Each half is a part of a rocker arm. The rest of each rocker arm forms part of the magnetic circuit of the rotor. So long as magnetic flux (established by the stator windings) is present, the halves of the roller nut are forced together and the lead screw of the control rod is engaged. If the magnetic flux collapses (as when the stator is de-energized) the halves of the roller nut are forced apart by springs and the action of the lead screw.

Such a motor requires a *special power supply.* The motor windings themselves are wound for the application of three-phase voltages which produce a rotating magnetic field similar to that produced in a conventional induction motor. However, one of the unique requirements of the power supply is that it be very low frequency, approximately 1 cps maximum. Another unique requirement is that when no motion of the rod is desired, the three-phase voltage cannot be simply turned off; to do so releases the rods and drops them into the core. The voltage therefore must be changed to direct current without changing its value. A third requirement is that the phase sequence of the three-phase voltage must be reversed, to reverse the direction of rod motion, without disconnecting the voltage.

This set of requirements is met by using a *low-frequency* DC-*to*-AC *mechanical inverter.* The inverter consists of a group of resistors connected in a continuous closed circuit (a closed ring) with commutator segments tapped to the junction points of the resistances. Diametrically opposite points on the ring of resistances are connected permanently to the terminals of a DC power supply. Three equally spaced brush arms rotate on the commutator formed by the taps. As the brushes rotate together, three-phase voltages are generated, the frequency of these voltages being determined by the speed of rotation and their phase sequence by the direction of rotation. Stopping rotation results in a DC output which holds the rod firmly in position.

The inverter is shown schematically in Fig. 9-7. Note that two sets of brushes are supplied and that each set supplies two rod motors. This feature has an important effect on the operation of the PWR core: any reactor control rod motion caused by energizing a small DC motor driving the brush arms must be the synchronized motion of four rods. By choosing the four rods connected to any one inverter so that there is one from each side of the seed array, core symmetry is maintained and problems of unequal distribution of neutron flux in the core are minimized.

The motion of the 8 normal inverters and 2 spare bus inverters controls the motion of the 32 control rods. The design data and design requirements used in developing the control circuitry for these inverters are described below.

(1) *Rod speeds must be adjustable* over sufficient range to meet uncertainties in rod worth, peak-to-average worth ratio and average-to-minimum worth ratio. Rod worth is approximately one percent reactivity per rod. The peak-to-average worth ratio is 2.85:1, and the average to minimum worth ratio is 5.0:1. Thus the system must allow for a range of control rod effectiveness of 14.25:1 over the operating range.

(2) To obtain a desirable axial flux shape, *rods must be divided into groups,* with the first group containing enough rods to allow initial criticality to

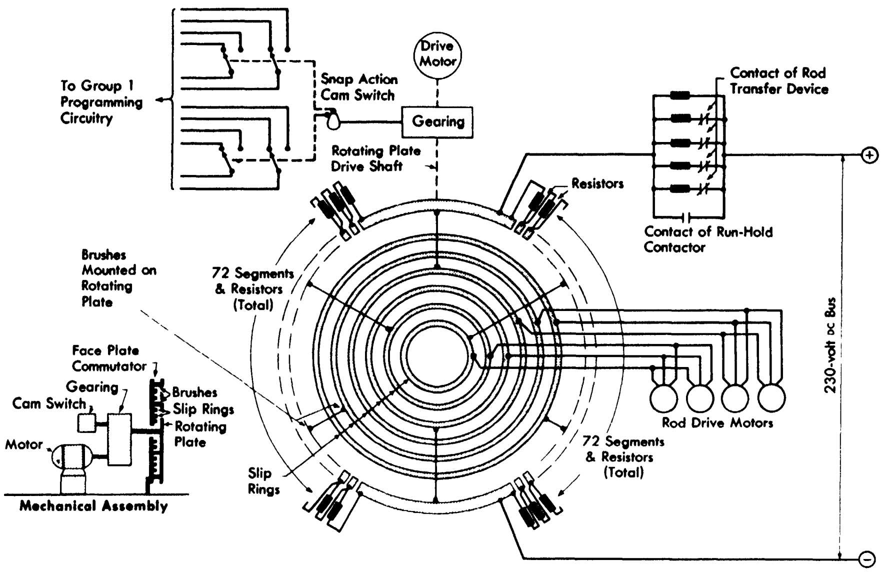


FIG. 9–7. Schematic diagram of rod control DC to AC power inverter.

be reached with this group approximately two-thirds out of the core when the core is new, clean, and at design temperature. The rods in this group are divided into symmetrical subgroups of four rods each, and these subgroups moved sequentially, with the maximum deviation of a particular subgroup from the remainder of the group being held to three inches. This was done to meet the requirement for maximum rate of reactivity change discussed in item (6) below.

(3) Initial core excess reactivity is such that the *first group must contain 16 rods.* The remaining 16 rods must be programmed as three separate groups of 8, 4, and 4 rods, in that order. Each group is programmed to the same height as the first group of 16 before the next group starts.

(4) The programming equipment must be sufficiently *flexible to allow programming* with 20, 24, 28, and 32 rods in the first group and the remaining rods, if any, grouped in multiples of four.

(5) Because rod worth is relatively low in the few inches near the top and bottom of the core, top and bottom *programming limits are established.* The lower limit is adjustable between 0 and 6 in. and the upper limit is adjustable between 52 and 70$\frac{3}{4}$ in. The upper limit is the point at which withdrawal of a group is stopped and that of the next group started when control has passed from the first group to subsequent groups of rods. The lower limit performs the same function for inward motion of rods under normal control. On scram, all rods drop to the bottom of the core. The initial settings of these lower and upper limits are 6 and 70 in., respectively.

(6) Under no condition except safety shutdown may the rod speed be permitted to exceed that which is equivalent to a reactivity rate of 7.3×10^{-4} Δk/sec. This rate has been established as the *maximum safe operating reactivity rate* based on a minimum of ten seconds being required to insert sufficient reactivity for prompt criticality.

(7) The rods must be *capable of following a xenon burnout* equivalent to a reactivity rate of 2×10^{-5} Δk/sec. Xenon transient studies indicate that this is the maximum xenon burnout rate to be encountered in normal plant operation.

(8) When a safety shutdown is required, all rods off the bottom must be *moved into the core by gravity*, regardless of the operating condition of the plant.

(9) A *safety insertion is required* for emergency conditions not warranting a safety shutdown. A safety insertion requires that reactivity be inserted at a rate equal to the normal operating rate, and the signal must override all manual or automatic rod control signals.

(10) The rod control circuitry shall be *locked in manual control* following a safety insertion.

(11) The *time to reach criticality should be as short as possible* without compromising the maximum withdrawal rate given above.

(12) The ability to move rods or to grant or withdraw permission for rod motion by others shall be *under the sole control of the reactor operator* at the reactor control console.

(13) Rod position indication for each rod with an *accuracy of* $\pm 1\frac{1}{2}$ *in.* is required for normal plant operation. Indication of rod motion which may be used by the reactor operator for incremental changes in reactivity is desirable. Rod position indication for each rod with an accuracy of $\pm\frac{1}{16}$ in. is required for test data.

With the requirements and rod worth data given above, values of linear rod speeds and startup times are determined as follows:

(1) The *maximum allowable rod speed* is taken as the speed required to insert reactivity for prompt criticality in 10 sec. It is assumed that 8 rods are moving simultaneously; this provides safety in the event that, through operator error, two groups of rods may be moved at once. The maximum speed is then obtained as follows:

$$\frac{\text{Max worth}}{\text{Rod-inch}} = \frac{\text{reactivity worth of rod} \times \text{max-to-avg worth ratio}}{\text{active seed length}}$$

$$= \frac{0.01\ \Delta k/\text{rod} \times 2.85}{70.75\ \text{in.}}$$

$$= \frac{4.03 \times 10^{-4}\ \Delta k}{\text{rod-inch}}.$$

$$\text{Max rod speed} = \frac{\text{max allowable reactivity rate}}{\text{no. rods moved} \times \text{max worth/rod-inch}}$$

$$= \frac{(7.3 \times 10^{-4} \times 60)\ \Delta k/\text{min}}{8 \times 4.03 \times 10^{-4}\ \Delta k/\text{rod-inch}} = 13.6\ \frac{\text{in}}{\text{min}}.$$

This is the maximum allowable rod speed under any operating condition except safety shutdown.

(2) The *operation rod speed,* or normal operating speed, is taken as the speed which will allow the worst expected xenon burnout to be overcome by rod motion in the region of minimum rod worth:

$$\frac{\text{Min worth}}{\text{Rod-inch}} = \frac{\text{reactivity worth of rod} \times \text{min-to-avg worth ratio}}{\text{active seed length}}$$

$$= \frac{0.01\ \Delta k/\text{rod} \times 0.2}{70.75\ \text{in.}} = \frac{2.82 \times 10^{-5}\ \Delta k}{\text{rod-inch}}.$$

$$\text{Operational rod speed} = \frac{\text{max xenon reactivity rate in } \Delta k/\text{min}}{\text{no. of rods moved} \times \text{min worth/rod-inch}}$$

$$= \frac{2 \times 10^{-5} \times 60}{4 \times 2.82 \times 10^{-5}} = 10.6\ \frac{\text{in}}{\text{min}}.$$

(3) Assuming that the core is new and clean, and that startup requires that 16 rods be withdrawn 60 in., the *minimum startup time* using the operational rod speed of 11 in/min is 22 min. This cannot be materially reduced without compromising the maximum withdrawal rate of 13.6 in/min given above. If xenon is present at startup, longer times will be required. Also, the time increases as the core ages.

Programming is accomplished by limit switches on the inverters and by limit magnetic amplifiers on the rod position indicators. Both limit controls determine the sequence of inverter operation in response to rod motion demands. The programming circuitry is designed so that any group of four rods may be utilized in any position in the order of withdrawing rods. The rod programming circuits also prevent inadvertent motion of more than one inverter (or of more than four rods) at a time. In addition, the electrical control circuits are designed so that circuitry failures will cause the inverters to stall rather than to reach excessive speeds.

Normal manual control of the rods is by a single switch on the control console which the operator may turn to demand either outward or inward rod motion by calling for the appropriate direction of rotation of the inverters. This signal is applied to one inverter at a time in accordance with the preset program. By means of two spare or alternate inverters to

which rods may be transferred, the operator at the console may adjust control rods independently of the normal program if necessary. The switches for these controls are shown in Fig. 9–5. The linear position of each rod is presented by a column of lights on the main instrument panel as shown in Fig. 9–6. The column of lights is operated directly from the secondary side of small transformers on the housing of the mechanism into which the control rod extension passes as the rod is withdrawn.

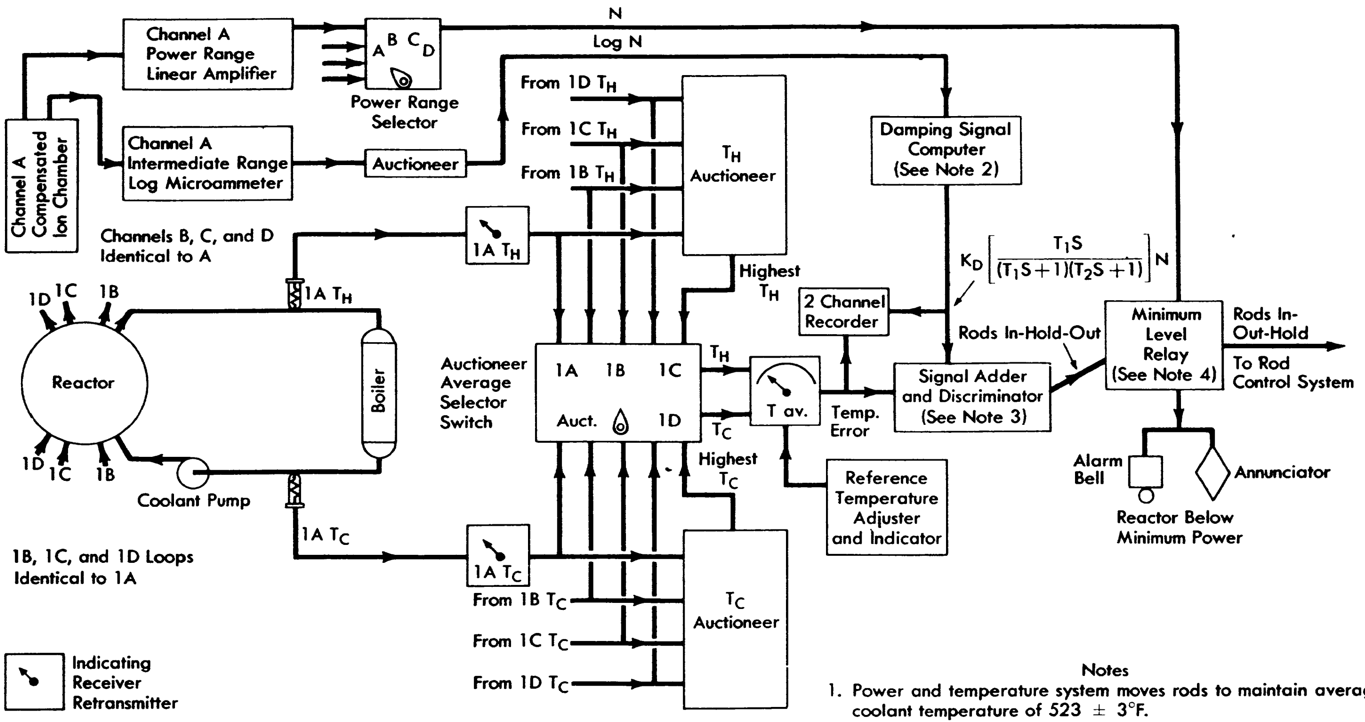
Reactor power and temperature control. The reactor power and temperature control system includes the equipment for automatic regulation of reactor power and reactor coolant temperature in the power operating range. Although the operator may control the reactor manually for all conditions, the automatic control system relieves the burden of constant operator attention to maintaining steady reactor coolant temperatures, especially during xenon variations.

Power control of the reactor is essentially automatic without the interposition of this external control circuit. This power control occurs through the effect of steam load changes in changing reactor coolant temperature, and the effect of the negative temperature coefficient of reactivity upon the reactor power level. The result is sufficient to make the reactor adjust to all normal load changes without moving the control rods. If it were not for the effect of the fission product poisons (primarily Xe^{135}) which change slowly over a period of hours after every reactor load change, no automatic control system would be needed. The operator would be required only to withdraw the rods for startup, regulate them to adjust plant temperature to normal value, and insert them for shutdown. Adjustment for fuel depletion would be gradual, requiring at the most only one or two motions per day. However, xenon poisoning does cause the negative temperature coefficient of the reactor to drift up or down on any load program except continuous steady power. To offset both the fission poisons and the temperature coefficient drift, control rod adjustments vary in frequency from two or three a minute to essentially none. Adjustments are made two or three times a minute following an increase in power to full load after an overnight shutdown; practically no adjustments are necessary if power is steady over long periods (of several hours duration).

To develop the automatic control system, the following system requirements were observed:

(1) The system must be able to operate over the power operating range of the plant, i.e., when the station generator is supplying power for either station service or the electrical distribution system. For practical purposes the system has been designed to operate at reactor power levels as low as 10% of full power. At 10% power (power decreasing) the reactor is automatically tripped to manual control.

(2) System design must be based on a temperature coefficient of reactivity of $-2.0 \times 10^{-4}\ \Delta k/°F$ and a pressure coefficient of reactivity of



Notes

1. Power and temperature system moves rods to maintain average coolant temperature of 523 ± 3°F.
2. $K_D = 0.5, T_1 = 0.1, T_2 = 20.$
3. Rods in $\Delta T = +3$; rods out $\Delta T = -3$.
4. Minimum level relay grants permission for automatic control at 10% and cuts out automatic control below 8% reactor power.

FIG. 9–8. Reactor power and temperature control system (simplified diagram).

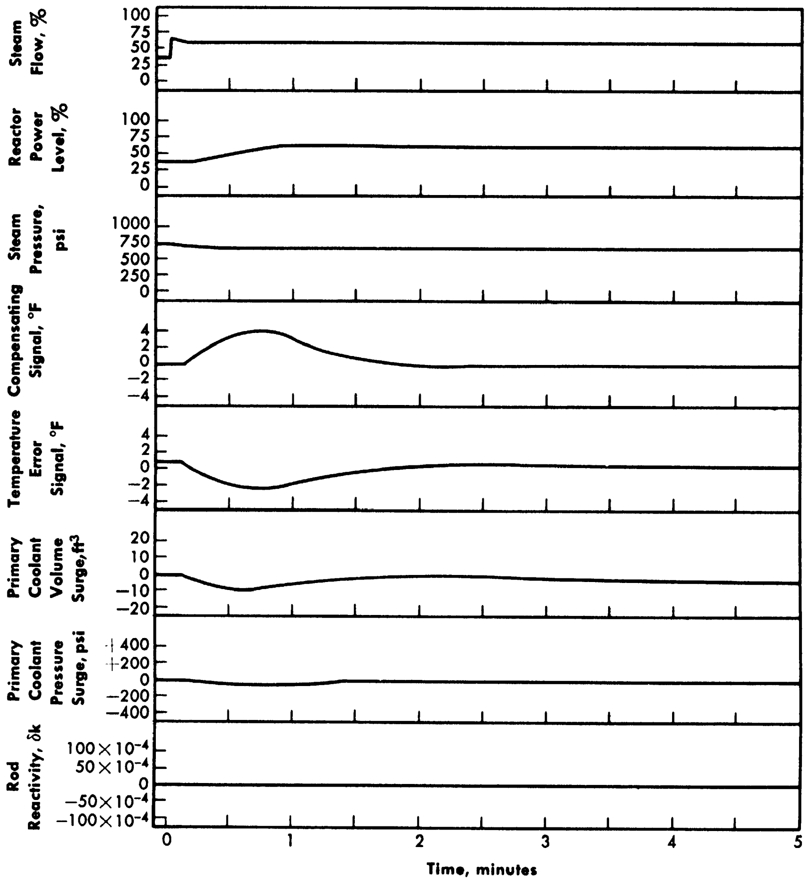


FIG. 9-9. Response of the plant and control system to an increasing load change. Steam flow change, 40% to 60% step; temperature coefficient, $-2 \times 10^{-4}\ \Delta k/°F$; control rod rate, $2.5 \times 10^{-4}\ \Delta k/\text{sec}$; reference temperature, 523°F.

$+2.4 \times 10^{-6}\ \Delta k/\text{psi}$. These are the minimum and maximum numerical values, respectively, expected for these coefficients.

(3) The system must provide stable operation at all operating power levels with rod speeds that permit the following of maximum xenon transients and with the ratio of maximum-to-minimum rod worth given in the preceding sections.

(4) The components of the system must be such that they provide accurate and reliable operation with long time stability. The average temperature computing equipment of this system must be able to reproduce input temperature signals within ±0.5°F.

The automatic control system is shown in schematic form in Fig. 9-8. It is, as previously noted, a temperature regulator using the temperature

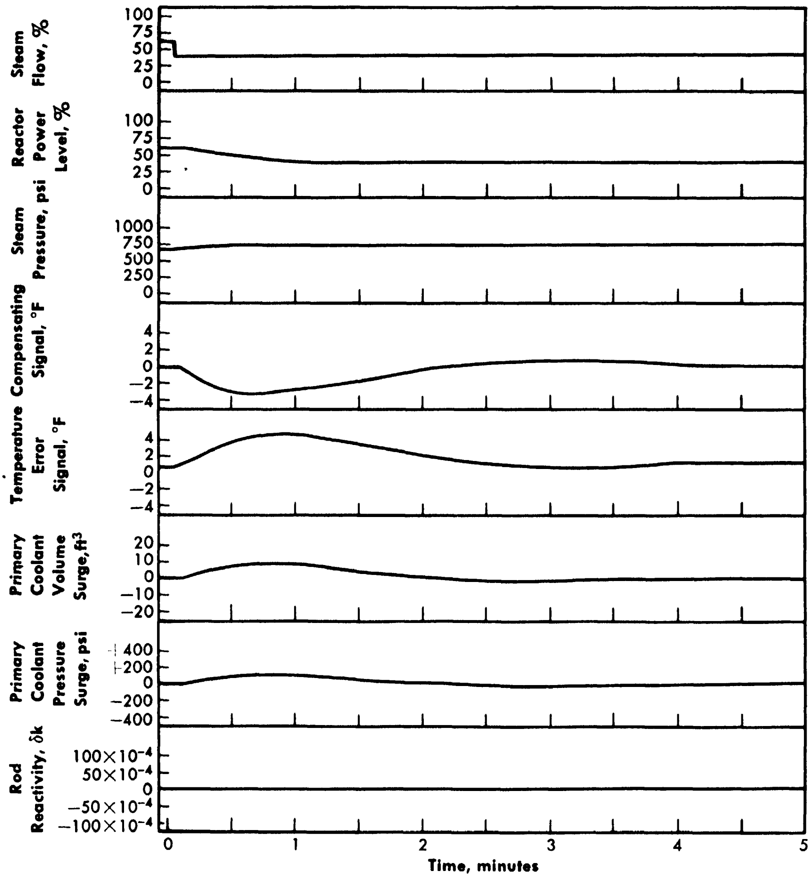


FIG. 9-10. Response of the plant and control system to a decreasing load change. Steam flow change, 60% to 40% step; temperature coefficient, $-2 \times 10^{-4}\,\Delta k/°F$; control rod rate, $2.5 \times 10^{-4}\,\Delta k$/sec; reference temperature, 523°F.

sensing elements described previously. A compensating signal derived from the rate of neutron flux change anticipates the appropriateness of corrective action, gives stability to the control system, and limits control action to that required for temperature changes produced by xenon poisoning.

The performance characteristics of the system are shown in Figs. 9-9, 9-10, and 9-11. The curves shown in these figures are from analog simulator studies of the dynamic response of the reactor plant (including the steam generators) with the control system connected. Preliminary operating data from the plant have confirmed the validity of these simulator studies.

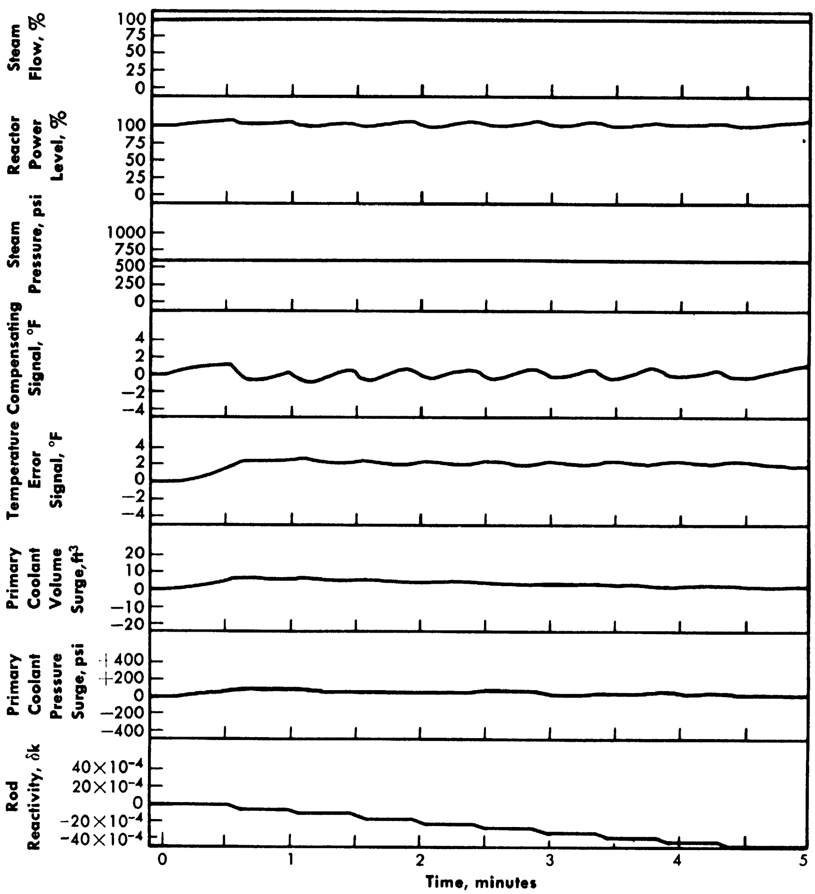


FIG. 9–11. Response of the plant and control system to a xenon burnout reactivity transient. Xenon reactivity rate, $2 \times 10^{-5}\,\Delta k$/sec; temperature coefficient, $-2 \times 10^{-4}\,\Delta k/°F$; control rod rate, $1.0 \times 10^{-4}\,\Delta k$/sec; reference temperature, 523°F.

Figure 9–9 shows the response of the plant and of the control system to a *step increase in steam demand* from a value corresponding to 40% reactor power to 60%. Note that although the coolant temperature changed by a total of $3\frac{1}{2}$°F during the transient, the compensating signal reflected the anticipated corrective action of the temperature coefficient in restoring temperature, and no control rod motion was demanded.

Figure 9–10 shows the response of the plant to a *transient in the reverse direction*, from 60 to 40% reactor power. Here also, the coolant temperature changed approximately $3\frac{1}{2}$°F but was corrected by the effect of the temperature coefficient of reactivity without control rod action.

Figure 9–11 shows the response of the regulating system and the plant to a *steady positive reactivity insertion* of $2 \times 10^{-5}\ \Delta k$/sec, a simulated transient of Xe^{135} reduction by reactor flux. The plant was at 100% power; the rate of reactivity insertion represents the fastest such transient expected as a result of xenon reduction. The control rods are inserted an increment each time the temperature error exceeds +3°F.

9–2.4 Reactor plant control systems. The vital plant parameters are coolant temperature, coolant flow, coolant pressure, and pressurizer coolant level. *Temperature* regulation has been described. *Coolant flow* is established by the four coolant pumps, which are operated at fixed speeds at all normal power levels. Control is manual; pump speed can be shifted to provide one-half flow during sustained low power operation or shutdown. The control switches for the coolant pumps are shown in Fig. 9–5.

Plant pressure is maintained by automatically regulated heaters in the pressurizer. Automatic action to restore pressure following large load increases is provided by a large block of heaters actuated by the temperature of the water in the pressurizer. Action to decrease pressure surges following large load reductions is provided by automatic on-off control of a spray valve in response to overpressure.

Coolant level in the pressurizer fluctuates with plant load because of temporary changes in bulk water average temperature. However, since the average of the coolant hot- and cold-leg temperatures is maintained constant in the steady state, there is very little permanent level change following a load change. The only requirement for level control is to make up water lost in drain operations or by relief valve operation, so only manual control of the drain valves and of the charging pumps is provided. These controls are located on the console shown in Fig. 9–5.

9–2.5 Reactor protection system. The reactor protection system protects any part of the reactor core from being subjected to temperatures which would result in core damage. The system senses those plant parameters which can give warning of a condition likely to result in an excessive core temperature and provides appropriate protective action.

Definition of accidents. Chapter 11 discusses the general problem of preventing core meltdown. Those accidents whose origin or prevention involves the reactor protection system are discussed in this section. These accidents are:

(1) Accidents involving loss of reactor coolant flow.
(2) Startup accidents.
(3) Cold water accidents.
(4) Accidents involving loss of reactor coolant pressure.
(5) Reactor control system (rod control and power and temperature control) failures.

A *loss-of-coolant-flow accident* is one in which the reactor coolant flow rate is suddenly reduced. The allowable neutron flux of the reactor is dependent upon the pressure and temperatures of the reactor coolant and upon its rate of flow through the core. Thus a loss-of-coolant-flow accident, by reducing the allowable flux, can result in the existing neutron flux exceeding the allowable neutron flux.

A *startup accident* is one in which the operator inserts an excessive amount of positive reactivity during withdrawal of the control rods to obtain criticality or during the rise in power from criticality to the power range. This results in a fast startup rate, with the neutron flux rapidly rising to an excessive level.

A *cold water accident* is one in which the temperature of the coolant entering the reactor suddenly decreases. This is effectively an insertion of positive reactivity which rapidly increases neutron flux.

A *loss-of-coolant-pressure accident* is one in which the reactor coolant pressure drops so fast that the saturation temperature of the coolant becomes less than the existing coolant temperatures. The boiling that occurs may seriously reduce the heat-removal capacity of the coolant.

General description of the system. The design criteria of the system pertaining to reliability are:

(1) The system must be fail-safe, so far as possible.

(2) The failure of any single component of the system should not shut down the reactor.

The types of protective action that the system provides may be summarized as follows:

(1) *Interlocks* that prevent the occurrence of certain accidents.

(2) *Alarms* that warn the operator of abnormal or potentially dangerous conditions.

(3) *Insertion* of the control rods, either partially or completely, at normal speed.

(4) *"Scram"* (release of the control rods), resulting in a fast and complete shutdown.

Simplified diagrams of the various portions of the reactor protection system are shown in Figs. 9–12 through 9–16 (see pp. 311–315). The diagrams illustrate, in a general manner, how the system utilizes input signals to provide the protective actions of alarms, safety insertion, and safety shutdown. The temperature interlock provision and system fail-safe features are not shown.

From the diagrams it can be seen that detectors or inputs (described in succeeding paragraphs) measure reactor plant conditions to obtain information for the system. In addition to these, detectors located in the station electrical output circuitry obtain information for the station electrical protection system and furnish signals to the reactor protection system. The various detectors, consisting of resistance thermometers,

pressure instruments, power relays, nuclear instruments, etc., occur in multiple in all cases. Their signals feed condition computers which are a grouping of detector signals of one type to obtain the information that an unsafe condition of a particular kind is present. As used here, the term "computer" may refer either to a grouping of contacts or to a bistable magnetic amplifier. The excessive neutron flux computer and the flow condition computer each feed two power-to-flow ratio comparators in parallel. One power-to-flow ratio comparator is used for safety insertion, the other for safety shutdown.

The computers and the power-to-flow ratio comparators feed four bistable magnetic amplifiers which increase the signal strength to provide power to operate the safety shutdown and safety insertion relays. The safety insertion amplifiers operate the insertion relays, consisting of two relays in parallel on the output side. These relays, which energize the rods-in contactors, always operate in parallel for any one condition requiring a safety insertion. The safety shutdown amplifiers operate the shutdown relays, consisting of two relays in parallel on the output side. These relays energize the shunt trip coils and de-energize the undervoltage trip coils of the circuit breakers supplying power to the rod drive motors. The safety shutdown relays and amplifiers are not in parallel on the input side. Different shutdown conditions cause their operation as discussed in following paragraphs. Removal of power from the stators of the rod drive motors immediately unlatches (25 msec) the rods, allowing them to fall into the core.

Summary description of system inputs. (1) *Four hot-leg resistance thermometer instruments* operate into a computer to produce alarm and shutdown action when coolant temperature in any one of the four loops leaving the reactor is excessive.

(2) *Three reactor coolant pressure instruments* operate through a low-pressure computer to provide annunciator and shutdown action when the pressure in either the pressurizer or the reactor vessel drops below a safe value. This computer may be blocked to permit cold startup. The blocking action may be removed manually as soon as the pressure is high enough to permit it and is automatically removed at 1800 psig. The blocking action is annunciated until removed.

(3) Each of *four neutron flux instruments* operates into an excessive neutron flux computer with five output levels for each instrument chain in the power range. For each of the five flux levels, the computer operates when two of the four instruments indicate the level is exceeded. The computer supplies two power-to-flow ratio comparators and also operates directly for the highest level to provide shutdown action when that level is exceeded. Calibrating signals are provided in the nuclear instrumentation to adjust and test all neutron flux trip settings.

(4) *Four pairs of valve position indicators* indicate to the loop status computer whether or not both main stop valves in each loop are fully open. *Four pump power relays* indicate to this same computer whether or not each of the four pumps is operating at greater than approximately 50% of its rated power for its selected speed. This computer operates through the shutdown amplifier to provide shutdown action when less than two of the four loops are in operation. By bias adjustment, this condition is set to be less than three of the four loops when reactor power is above 44%.

(5) The loop *main valves and the pump power relays* also indicate the reactor plant operating condition to three flow condition computers which set limits of power operation. In addition, the flow condition computers sense that a pump circuit is set up for one-half speed. One flow condition computer trips when at least one of the four loops is not in full operation, as determined by the power input to the pumps, the valve position, and the speed selection. A second flow condition computer trips when at least two of the four loops are not in full operation as determined above. A third flow condition computer trips when at least three of the four loops are not in full operation as determined above. It should be noted that while this third computer permits operation on one loop at full pump speed, the loop status detector bars this operation. The detector establishes neutron flux trip points for startup of the plant with pumps at one-half speed.

(6) The *power-to-flow ratio comparators* detect an excessive neutron flux for the existing flow condition. One operates through amplifiers to drive the insertion relays in order to reduce the neutron flux and thus correct the excessive power-to-flow ratio. The other operates through a single amplifier to drive a shutdown relay, releasing the control rods.

Electrical fault condition detectors in the generator, main transformer, and generator-lead station service transformer protection systems provide a signal to the safety insertion relays to shut down the reactor at the same time that they shut down the turbine and generator.

(7) The *manual shutdown signal* directly removes power to the control rods. When a shutdown occurs, it is annunciated and the turbine throttle is tripped. When an insertion occurs, it is annunciated and the power and temperature control system is tripped out of automatic control.

(8) *Startup rate limitation* is provided from the intermediate range of the nuclear instrumentation by the excessive startup rate computer which receives startup rate signals from the four neutron flux instrumentation chains. This computer operates through a startup rate lockout, which is a manual switch. The lockout is used to delete the startup rate limitation signal during power operation. The computer trips the insertion amplifier when two of the four channels of information indicate excessive startup

rate. The protection must be reinstated to energize the source range instruments when beginning a startup rod withdrawal. It should be noted that the source range instrumentation may also be used to provide a startup rate limit signal. The signal from this range is not connected because it is not considered necessary for protection and is apt to produce spurious signals at very low levels.

A summary of all system inputs and the protection provided by each is shown in Table 9–1 (see pp. 320–324). This table shows alarm functions as well as insertion and shutdown functions. The specific conditions for which protective action (insertion or shutdown) is taken are summarized as follows and are described in detail in subsequent articles.

(1) Excessive reactor coolant hot-leg temperature.
(2) Reactor startup rate faster than the maximum allowable value.
(3) Low reactor coolant system pressure.
(4) Less than two of the four reactor coolant loops in use, or less than three if reactor power is above 44%.
(5) Excessive neutron flux for the existing reactor coolant flow rate.
(6) Excessive neutron flux under any condition.
(7) Loss of the AC or DC power to the system which would otherwise prevent protective actions from being taken.
(8) Generator shutdown by the operation of the generator protection system.
(9) Temperature of the water in an inactive steam generator sufficiently lower than the highest of the cold-leg temperatures of the coolant loops in use if the loop containing that steam generator is to be placed in service.
(10) Operator signal for a scram.
(11) Activation of decay heat removal system.

The delay times of the system for shutdown under various conditions are shown in Figs. 9–17 through 9–22 (see pp. 316–319). These times are designed to provide adequate reactor protection under accident conditions.

Protection against excessive reactor coolant temperature. The sensing elements for reactor coolant temperature protection are four resistance thermometers, one in each loop, near and downstream of the main stop valves, in the main coolant piping leaving the reactor vessel.

The transit time of the water at full pump speed from the top of the core to the resistance thermometer is approximately 4.5 sec. The response time of the thermometer is approximately 3 sec. The sensing elements are connected to servo null balance receiver units; these units, located in the control room, are driven by magnetic amplifiers and have adjustable set points for high T_h alarm and high T_h shutdown. The four contacts for alarm and the four for shutdown are connected in parallel for each condition, providing an alarm or a shutdown for any one of the four units

reaching the appropriate set point. The contacts for shutdown directly energize the two shutdown relays.

A hot-leg temperature that becomes abnormally high, and is not caused by a sudden reduction of flow or a rapidly increasing neutron flux, may be assumed to be a slowly rising temperature. The time response of the T_h sensing elements is slow compared with the time response of the neutron flux measuring instruments and the time response of the means of detecting flow reduction. Therefore T_h thermometers are not used as prime sources of protection involving these transients; they do provide some backup protection, but not enough to prevent core damage from a severe neutron flux transient or coolant flow transient otherwise unprotected against.

Causes of high T_h for which the hot-leg thermometer can give protection are failures in the control system which would: (1) cause the control rods to move outward, thereby inserting reactivity at their maximum or lesser worth, (2) prevent the control rods from compensating for fast xenon transients.

Thus the principal protection that can be provided by use of the hot-leg thermometers is to prevent local boiling under steady-state or relatively slow transient conditions. The core is designed to have an operating hot spot metal surface temperature of not more than 636°F when producing maximum power for the existing flow rate. From the Jens and Lottes equation for the prediction of local boiling, local boiling will occur, at maximum power and a coolant pressure of 2000 psia, when the hot spot temperature reaches approximately 642°F. By indirectly monitoring the average temperature of the coolant leaving the core, hot-leg thermometers can indicate when local boiling is likely to occur.

Design is based on a nominal average temperature of the coolant of 523°F and a coolant pressure of 2000 psia. Design tolerances on these values are ± 5°F for temperatures (± 3°F for control dead band and ± 2°F for instrument error), and ± 30 psi instrument error for pressure.

Shutdown action is provided for T_h greater than 550°F for the following reasons:

(1) Local boiling will be essentially prevented in the steady state, providing the coolant pressure is maintained at 2000 psia.

(2) The control rods are released instead of being inserted, since a component malfunction in the rod control system would conceivably be the reason that the hot-leg temperatures have become excessive.

(3) If the average temperature of the coolant is maintained at 523°F, the hot-leg temperatures will not exceed 550°F as a result of any loss-of-flow accident except one so severe that bulk boiling is likely to occur. In this case, the control rods would already have been released by the high nuclear level or loss-of-flow protection so as to prevent bulk boiling.

The alarm limit for T_h is set at 544°F. With this setting, minor primary system transients will not cause alarm. The setting is low enough so that if temperature control were not being maintained and a maximum xenon burnout were occurring, the operator would have time enough after the alarm to halt the upward drift of coolant temperatures by manual operation and thus prevent the release of the control rods.

Protection against excessive startup rate. Protection against excessive startup rate is provided by the information received from the intermediate range of nuclear instrumentation. Four channels of information furnish rate of neutron multiplication in decades per minute. When this rate exceeds 1.74 decades/min in any two of the four channels, the startup rate computer provides a signal to the insertion amplifiers. The insertion signal is automatically removed when the rate signal drops to approximately 1.2 decades/min. Provision is made to permit the requirement for coincidence to be changed to single channel (any one of four). This provision allows emergency operation with one channel of instrumentation out of service without reducing the degree of protection.

The startup rate detectors protect against the excessive neutron flux level and core damage that will result from the reactor power entering the power range while increasing at such a rapid rate that the negative temperature coefficient of reactivity cannot limit the maximum level reached to a value less than that which would cause core damage. The startup rate limit, if it is to be the primary source of protection for startup accidents, should provide a turndown in power level before the bulk boiling limit is reached. If it does not, the power-to-flow ratio protection will effect a shut down.

Fast startup rate accidents are caused by:

(1) Excessive withdrawal of the control rods when obtaining criticality.

(2) Continued withdrawal of the control rods beyond the point that provides a proper rise in the neutron flux level, while the reactor is still operating below that value of flux at which coolant temperature is significantly affected by small flux variations.

(3) Accidental opening of the steam throttle valve when the reactor is operating in the condition described in (2) above during a hot startup.

(4) Accidental startup of a cold coolant loop when the reactor is operating in the condition described in (2) above during a hot startup.

An insertion, rather than a release, of control rods is provided because:

(1) The first two causes of startup accidents listed above are operator carelessness or error, and it is necessary only to take control away from him. For this purpose, a stopping of the rods would be sufficient.

(2) The third cause of startup accidents listed above would produce a fast startup rate with no rod motion. Hence stopping the rods would have no effect on limiting the rate of rise of neutron flux to the power range.

However, the minimum time in which the throttle can be fully opened by use of the speed changer control is approximately 33 sec. Because of this slow rate, there is ample warning of improper action in the response of the temperature and pressure instruments, and time to take action to close the throttle. Hence an insertion is provided to prevent the startup rate from becoming excessive.

(3) Startup accidents due to the fourth cause are prevented by action of the loop temperature interlock. This bars starting the main coolant loop pumps until the out-of-service loop temperature is within 20°F of the in-service loop temperatures. A normal rod insertion speed will reduce the startup rate to a low value if the accident occurs at the beginning of the startup range. If it occurs when the reactor is almost at the power range, the negative temperature coefficient of reactivity will prevent an excessive rise in power.

Protection against low coolant pressure. Protection against low coolant pressure is provided by pressure sensing instruments in the pressurizer and in relief valve piping directly connected to the reactor vessel. A narrow range pressure instrument in the reactor plant pressurizer supplies the closed contacts to indicate (on the station annunciator) an alarm pressure at 1850 psig. A wide range pressure instrument in the reactor plant pressurizer and a wide range pressure instrument on the reactor vessel relief valve line provide the closed contacts initiating safety shutdown when the coolant pressure drops to 1600 psig.

The annunciated limit is set at a value lower than the pressures expected to be encountered as a result of operating transients. Annunciator action for pressure less than this limit calls for operator evaluation and then operator action. The low pressure may be a transient condition resulting from a larger than normal load increase and may require no action, or it may be due to a coolant system leak that can be isolated, or a pressurizer heater failure that can be corrected. Pressures less than 1600 psig should be encountered only in the event of a serious coolant system accident. Release of the control rods is provided for pressures less than this limit to avoid bulk boiling in the steady state and to give the fastest action possible in the event of a serious rupture in the reactor coolant and reactor coolant auxiliary systems.

Since the reactor may be started up with the coolant cold and its pressure below 1600 psig, means are provided for removing the low-pressure shutdown feature during such operation. If the reactor plant is being cooled down with the reactor operating, there is no automatic removal of low-pressure protection. The operator must do this manually, or encounter a shutdown when 1600 psig is reached. While the protection is removed, this fact is indicated on the station annunciator. When the reactor coolant pressure is being increased by reactor heat, the low-pressure protection

may be restored manually above 1600 psi. It is automatically restored when the pressure reaches 1800 psi.

Protection against low reactor coolant flow. The system "sees" a reactor coolant loop as being in use when both main valves in the loop are fully open, the pump is operating, and the input electrical power to the pump is greater than approximately 50% of its rated input for the speed at which it is operating. The system will interpret less than two loops in use as a severe, or potentially severe, loss-of-flow accident and will provide shutdown action without reference to neutron flux level.

So that the system will not respond to transmission line transients which temporarily reduce the power to the pumps, a contact on each pump power relay energizes an adjustable timing relay. On a reduction or loss of pump power, the pump power relay de-energizes the timing relay. If the pump power relay has not started to pick up when the timing relay has completely dropped out, then the loss of power is not considered to be a fault transient; the appropriate control winding of the loop status computer is energized, in recognition of the fact that that loop is out of service.

Considerations made in determining the trip setting of the pump power relays were:

(1) By making the trip setting close to the normal amount of electrical power the pump requires, a greater proportion of pump malfunctions are quickly detected. However, if the trip setting is too close to this normal input power, the relay will be overly sensitive to the extent that it might operate for relatively minor power fluctuations.

(2) The speed of response of the pump power relays to *both* complete loss and complete restoral of power should be as fast as possible consistent with good reliability. Having the trip setting close to normal input power tends to lengthen reset time or the speed of response to complete restoral of power. If the setting is close to zero power, then the dropout time or speed of response to a complete loss of power will tend to be longer.

(3) Although the pumps have two operating speeds, the trip setting may be accurately adjusted only for one speed. The trip setting will then be only approximately the same for the other speed.

In view of these considerations, a trip setting of approximately 50% of rated power at either speed of operation has been selected.

A sudden and complete loss of electrical power to a pump is actually sensed in about 7 to 10 msec or 0.4 to 0.6 cycle (60 cps basis) by the pump power relay. The timing relay is delayed for an additional 8 cycles to allow for clearing transmission line faults and restoring power. Thus approximately 9 cycles, or 150 msec, are obtained to ensure that the loss of power is not an electrical system fault transient. Since 2 cycles are needed to reset the pump power and timing relays, the timing relay will

be prevented from dropping out if the power is restored within 7 cycles after its loss. The relays which sense transmission line faults operate in approximately 1 to 2 cycles. The circuit breakers have a 3-cycle interrupting time. Allowing 1 cycle for voltage restoration (voltage rise after fault removal), a total time of 5 to 6 cycles is required for line fault clearing. Since this is within the permissible 7-cycle period, the timing relays do not trip out on such faults.

It is obvious from the preceding discussion that considerable dependence is placed on very rapid clearing of transmission line faults. Should a delay of 1 to 2 cycles (17 to 34 msec) occur in line fault clearing, the system will take action to shut down the reactor. This close relaying is required, however, to prevent damage to the core during loss of coolant flow.

A severe loss-of-flow accident due to the simultaneous or near simultaneous loss of three or more operating pumps will most probably be the result of loss of electrical power to the pumps. The loss of power is sensed in approximately 150 msec, well before the flow coast-down is completed. That is, the inertia of the water and the pumps delays the loss of flow sufficiently so that, in general, release of the control rods will prevent bulk boiling. The exception to this is certain sequential loss-of-flow accidents where the final and major loss of flow occurs when temperatures in the core are already abnormally high. These sequential accidents result in compounded transients and require more stringent time response requirements of a protection system than do the simultaneous total loss-of-flow accidents.

It can be seen from the discussion in the preceding paragraph that little opportunity exists for providing action that is faster than that already provided. Although the approximately 8-cycle delay could be eliminated so far as circuitry is concerned, it would be at the expense of plant reliability in that the plant would be shut down when transmission line faults occur. To provide protection against the sequential loss-of-flow accident, it has been necessary to rule it out of existence, essentially. This is done by automatically biasing the loop status computer so that it will trip for less than three loops in operation if reactor power is above 44% of capability. Thus it effectively rules out of existence all sequential loss-of-flow accidents except the one in which one pump out of four operating is lost, followed by the loss of the remaining three pumps. Calculations show this sequence of events to be tolerable with the present time response.

During plant startup, when reactor power is low, protection must be based on shutdown for less than two loops in operation to permit deenergizing coolant pumps during speed changing and during 2400-volt bus transfers, a normal part of the startup procedure. There is no problem resulting from a sequential loss-of-flow accident if the reactor power level is less than 44% of three-loop rated capacity.

From an operating reliability point of view, it would be desirable to have the reactor tolerate, without damage, the loss of at least one coolant pump when three are operating or two if four are operating. It is not certain that these losses of flow are not tolerable. In the face of uncertainty, system design has been conservative until it becomes possible to prove that the conservatism is unwarranted. During the first year of operation, extensive tests will be performed to determine how severe the transients are from the point of view of producing excessive core temperatures. These tests may show that the protective features can be relaxed.

Protection against excessive neutron flux. Neutron flux linear level information is received from four independent channels of nuclear instrumentation and is supplied by the nuclear instrumentation system. The information is in the form of a current, the magnitude of which is proportional to neutron flux. For each channel, the output of the linear level amplifier of the nuclear instrumentation system is connected to control windings on five bistable magnetic amplifiers, each of which is set to fire at prescribed flux levels (see Figs. 9–12, 9–13, 9–14).

For a given set point, the bistable magnetic amplifier of each of the four channels energizes one control winding of a nuclear coincidence magnetic amplifier, when the set point is exceeded. When any two of the control windings have been energized, the coincidence magnetic amplifier fires, and initiates shutdown action either directly (if it is the one corresponding to the highest neutron flux limit), or in conjunction with flow condition magnetic amplifiers (if it is one of those associated with a lower set point).

Backup protection is the primary function of excessive neutron flux protection for any steady flow condition. For each of the two normally permitted flow conditions (three-loop and four-loop flow) the maximum transient neutron flux limit is set at 138% of the reactor capability for that flow. A neutron flux which rises to and exceeds the transient limit for a flow condition may be caused by a startup accident or by a rapid drop in temperature of the coolant entering the reactor.

The primary protection for a startup accident is the startup rate protection discussed previously. That is, with a 1.74 decade/min startup rate limit, it is expected that this protection (startup limit) would prevent the nuclear level from reaching the transient limit. The primary protection against cold water accidents is prevention through the medium of the ΔT interlock discussed below. The accidents which would result in slowly rising neutron flux to the transient limit are operator error (excessive control rod withdrawal in the power range), and plant overload by throttle opening or relief valve malfunction. It is expected that excessive hot-leg temperature protection or steady-state limit protection in which a power cutback occurs will govern in these cases. Power cutback is discussed in the next section.

Protection against excessive power-to-flow ratio. An excessive power-to-flow ratio is an excessive neutron flux for the existing coolant flow. In establishing that the condition exists, power-to-flow ratio comparators are used to indicate when the neutron flux exceeds the *steady-state* limit and when it exceeds the shutdown (*transient*) limit for each of four flow conditions. The power-to-flow ratio comparators indicate excessive power-to-flow ratios by the firing of a magnetic amplifier when both of the control windings of the particular amplifier are energized. Energizing of one of the control windings indicates the existence of a given flow condition; energizing of the other control windings indicates an excessive neutron flux for that flow condition.

The four flow conditions are defined and are established by the circuitry as follows:

Flow condition No. 1: All four coolant pumps operating at normal power input for their speed condition.
All four coolant pump speed selector switches set for full speed.
Both main loop hydraulically operated stop valves in all four loops fully open.

Flow condition No. 2: All the above conditions fulfilled for any three of the loops.

Flow condition No. 3: All the above conditions fulfilled for any two of the loops.

Flow condition No. 4: The above conditions with respect to coolant pumps and loop valves fulfilled for two, three, or four loops, except that their associated speed selector switches are in the one-half speed positions.

The four flow conditions are established by bistable magnetic amplifiers which have their control windings energized by the actions of pump electrical power relays or by valve position relays. Since flow condition No. 1, as defined above, represents maximum possible flow, no magnetic amplifier is provided for this condition. The neutron flux limits for power cutback and for shutdown operate directly on the system output relays. Thus they are always in effect even though there may be other (lower) limits established by the existence of a lower flow condition.

The rated neutron flux level or capability of the reactor is proportional to the rate of reactor coolant flow. The reactor is normally to be operated within its rated level for the existing flow rate, and this is the responsibility of the operator. Transient excursions above the rated level are permissible provided the transient limit is not exceeded. The theoretical transient limit is that neutron flux level for a given flow rate and coolant temperature at the core inlet that causes bulk boiling at the exit of the hot channel

to begin. The transient limit is a function of the coolant pressure and the amount of local boiling in the hot channel. To determine the neutron flux level set point that will prevent an excursion beyond the transient limit, consideration must be given to the type of accident and the amount of overshoot before a downward change in flux begins. *Hence the set point is lower than the theoretical transient limit.*

In general, for those accidents in which the rated neutron flux level is likely to be exceeded and in which the transient limit could be exceeded, the transient limit will be about 140% of rated level provided that the coolant pressure is at least 2000 psia and the average temperature of the coolant is not more than 523°F when the accident occurs. To allow for short-time excursions above the rated capability of the core to permit the generating unit to absorb its share of disturbances caused by losses in generating capacity elsewhere in the Duquesne Light Company transmission system, a steady state is defined. The Shippingport Station share of such disturbances would not exceed 10 to 15% of its rating based on the loss of the largest unit of the transmission system. The station is not required to operate between the steady-state limit and the transient limit for more than the length of time it takes to cut power back by automatic rod control action. Nor should it be allowed to operate in this region, for to do so makes the core subject to bulk boiling before scram action can take place if a loss of reactor coolant flow should subsequently occur.

Thus a region between the steady-state and transient limits for each flow condition is defined as one in which automatic power cutback by normal insertion of the rods takes place until the neutron flux level is below the steady state limit. Between the steady-state limit and the rated capability for each flow condition, no automatic action is taken, but the operator will take action to reduce load to rated value as soon as system disturbances such as those described above can be absorbed by other less heavily loaded units on the transmission system. This time is determined by the automatic load control system and would not exceed approximately two minutes. The steady-state limit is established at approximately 115% of rated capability for flow conditions representing three and four loops in operation.

The neutron flux values for steady-state and transient limit set points are established on the basis of Table 9–2, which shows coolant flows for the four flow conditions, three-loop flow being taken as 100%. The reactor capability for flow condition No. 4 is based on two-loop, one-half speed flow because this condition is possible. Actually, three and four loops in operation with the pumps at one-half speed are also included in flow condition No. 4. Although it is not expected that the reactor would ever be started up with only two loops operable, this possible condition is taken as the limiting value.

Table 9–2

Flow Percentages for Four Flow Conditions

Flow condition	Coolant flow, %
No. 1—Four loops, full pump speed	125
No. 2—Three loops, full pump speed	100
No. 3—Two loops, full pump speed	70
No. 4—Two loops, one-half pump speed	35 (minimum)

Table 9–3

Flow conditions as defined in Table 9–2	No. 1	No. 2	No. 3	No. 4
Percent of reactor power for safety insertion (cutback)	138	114	61	44
Percent of reactor power for safety shutdown (scram)	175	138	114	61

The summary of protective action for excessive power-to-flow ratio is as shown in Table 9–3.

Protection against loss of control power. There are two sources of control power for the reactor and reactor plant. Alternating-current control power is used for nuclear instrumentation, for reactor plant instrumentation, and for that part of the protection system which operates on magnetic amplifier circuitry. Direct-current control power is used for the reactivity control system (operation of the drive motor inverter power supplies and the programming of them, and for closing and tripping the safety shutdown and motor-generator set circuit breakers) and for that part of the protection system which utilizes instrument contacts to directly trip the shutdown relays. Regulated, rectified DC control voltage is used in the magnetic amplifier bias supplies.

(1) *Alternating-current control power* is obtained from a 480 to 120-volt transformer connected to one of the 480-volt reactor plant buses. On loss of this voltage, the control power is transferred in 50 msec to a second 480 to 120-volt transformer energized from a second 480-volt bus. The two buses are fed from separate 2400-volt sources, one of which is normally supplied by the main generator, the other normally connected to the 138-kv transmission system.

(2) *Direct current control power* is obtained from a 125-volt control battery. Main circuits of this control power supply may be manually

transferred to the station emergency control battery by a fast-acting switch.

The protection system is fail-safe for loss of control power, since such loss of power would make the system inoperative for conditions requiring safety shutdown. Such fail-safe features must be sufficiently slow to prevent an unwarranted shutdown because of station power supply bus transfers or momentary (9 cycles or 150 msec) reduction of station voltage to low values during transmission line faults.

Loss of DC control power to that portion of the reactivity control system concerned with the rod power supply circuit breakers and of DC control power to the safety shutdown relays results in shutdown in 1.0 sec. This delay allows transfer of the circuit to the emergency battery. These functions are supplied by one feeder from the reactor plant control battery. The transfer action results from the action of the undervoltage trip coils of the rod motor power supply circuit breakers.

The four channels of nuclear instruments are supplied with four separate feeders from the AC control power bus. Shutdown in 0.25 sec. is provided on loss of voltage to any two of these four feeders.

The flow condition, loop status, neutron flux level coincidence, and power-to-flow ratio comparator magnetic amplifiers cannot be divided into separate groups for power supply. Hence a single 120-volt AC feeder is used to supply this part of the system. The DC voltage for bias supply is taken from this feeder so that independent voltage variations will not produce false action in the magnetic amplifiers. Failure of this AC feeder would prevent all shutdown action with the exception of low pressure and high temperature. Hence its failure causes shutdown, which occurs in 0.25 sec.

Such a time delay of 15 cycles is used for loss of AC voltage to ensure adequate protection for the total loss-of-flow accident. In the implausible event that a total loss-of-coolant flow accident should occur, it could possibly be accompanied by a loss of AC control power. Such a loss would prevent action of the system through the power-to-flow ratio circuitry. The undervoltage protection, set at 0.25 sec. delay, would produce shutdown in approximately 0.35 sec. The permissible time delay for a loss of four pumps is 0.58 sec. and for a loss of three pumps (three initially operating) is 0.62 sec.

Interconnected shutdown of turbogenerator and reactor plants. When the generator is shut down by the protection systems on the generator, main transformer, or generator-lead station service transformer, it is expected that the plant will remain shut down long enough to warrant shutting down the reactor. In such cases the reactor is shut down by inserting the control rods at normal speed.

Such a generator shutdown can result from any one of the following:

(1) Generator differential relay trip.
(2) Main transformer differential relay trip.
(3) Sudden pressure relay trip on the main transformer.
(4) Station service transformer differential relay trip.
(5) Sudden pressure relay trip on the station service transformer.

Turbine throttle trips caused by mechanical troubles in the turbine portion of the plant are not cause for reactor shutdown. Such troubles may be transitory, with the possibility that the operator can recover the station.

When the reactor is shut down by a release of the control rods (scram), steam flow from the boilers must be reduced as quickly as possible to prevent excessive heat withdrawal from the reactor coolant system and consequent thermal shock to the system. Thus, a contact on each of the safety shutdown relays provides a closed contact signal to the turbine protection system to trip the turbine throttle.

Cold water accident prevention. A cold water accident would be caused by placing a loop in service when its temperature is appreciably lower than that of the operating loops. The primary protection, preventive in nature, is provided by the temperature difference interlock as described previously in this article under "Protection against excessive startup rate."

Manual safety shutdown. Manual safety shutdown of the reactor at the operator's discretion is provided by the safety shutdown switch on the reactor section of the main control console. It can be seen in the picture of the control panels (Fig. 9-5) at the top center of the reactor symbol area.

Activation of decay heat removal system. An auxiliary function of the reactor protection system is to provide means for automatic actuation of the decay heat removal system. This is accomplished by a relief valve on the main steam system which opens automatically when the steam pressure is above 707 psi. The decay heat removal valve is actuated if all coolant pumps are de-energized, thus stopping forced flow through the core. Note that automatic actuation of this valve is not provided for all scrams. So long as there is forced circulation of coolant through the system, heatup from decay heat is slow and the operator can regulate the valve manually to hold temperatures to normal values.

System annunciation. Plant annunciation informs the operator of certain actions of the system and of certain conditions within the system. To provide annunciation, the system furnishes various signals in the form of closed contacts. Should the signal from the system be removed, annunciation continues until acknowledged by the operator.

The annunciator alarms that are considered to be a part of reactor protection are as follows:

(1) Safety insertion (cutback) occurring.
(2) Safety shutdown (scram) already occurred.
(3) Loss of flow in any loop (four separate annunciators).
(4) Low reactor coolant pressure.
(5) High reactor coolant pressure.
(6) High reactor coolant temperature.
(7) Low-pressure protection removed by use of the cutout switch.
(8) A reactor protection system test switch actuated without permissive action of the reactor operator.

System test provisions. The magnetic amplifiers of the system normally operate de-energized, or not firing. To have them normally operate fired would make the system subject to shutdown action as a result of momentary voltage interruptions caused by station service system bus transfer or by transmission system line faults. It would also make the system subject to shutdown action as a result of a number of single component failures. Although magnetic amplifiers are considered to be extremely reliable, failures are possible. In general, failures in bistable magnetic amplifiers are not detectable without actually testing them to determine that they function when the appropriate signals are applied.

The principal function of the test equipment is to provide a means by which the integrity of the system may be determined at periodic intervals by applying simulated signals. This function can be carried out with the reactor in operation. The equipment is arranged so that the output circuits are interrupted momentarily while the test signal is applied, thus preventing reactor shutdown during testing. A further function of the test equipment is to enable the operating personnel to determine trip settings of protective equipment.

The following criteria were applied in the design of the test equipment and the test procedure:

(1) The functional test procedure should be capable of being carried out during reactor operation.

(2) The functional test procedure should be capable of providing pre-startup assurance of equipment integrity.

(3) All equipment in the reactor protection system (or equipment located in other systems, but which is a part of reactor protection) which cannot be readily monitored visually during reactor operation should be subject to checkout by the test equipment.

(4) When the system is tested during operation, protection must not be completely removed from the reactor.

(5) Failure of the test equipment should not disable the system.

(6) Failure of the test equipment should not cause a shutdown.

(7) Operator errors in following the test procedure should not cause a shutdown.

(8) It is desirable not to have the test procedure carried out by the reactor operator, so that he may devote full attention to conditions of the reactor and reactor plant during testing. However, test permission and control of the test procedure should be under this operator's jurisdiction insofar as possible.

(9) The test procedure should be simple and require minimum time to execute.

(10) The test procedure should not require changes in station operation (such as load variation) to obtain a satisfactory test result.

9-2.6 Radiation monitoring. Radiation monitoring has four main functions: (1) *to warn the operator* of defects in the reactor coolant systems which might not be detected by other means, (2) *to measure radiation levels* in various areas to control personnel movement, (3) *to monitor the processing* of the radioactive wastes, and (4) *to furnish historical data* on the effects of the Shippingport plant, if any, on activity levels in the surrounding area.

The *first function* is accomplished by detectors which sample the air in the various plant container compartments and in the ventilation exhaust stack, and by beta-gamma detectors which monitor the activity of a sample of secondary water from each of the steam generators. The detectors in the ventilation exhaust stack set off an alarm for excessive radioactive airborne gaseous and particulate matters, and also have a control function—they will cause the container ventilation valves to close if the radiation level of the exhaust air exceeds a value which is one-tenth of handbook tolerance.* The beta-gamma detectors warn of leaks between the reactor coolant and secondary systems in the heat exchangers which would cause the steam to the turbine to become radioactive. These detectors actuate alarms both in the main control room and in the turbine room basement. Both types of radiation monitors can detect activity from small coolant leaks before the leaks become noticeable by other means.

The *second function* of radiation monitoring is accomplished primarily by beta-gamma detectors in the plant containers. These are located in the various access areas of the containers to provide remote readings before access is permitted. Other beta-gamma count rate detectors are placed throughout the normally occupied areas of the station.

The *third radiation monitoring function* is accomplished by detectors on the various decay tanks of the radioactive waste disposal system, by

* National Bureau of Standards Handbook 52, "Maximum Permissible Amounts of Radioisotopes in the Human Body and Maximum Permissible Concentration in Air and Water"; National Bureau of Standards Handbook 61, "Regulation of Radiation Exposure by Legislative Means"; and Pennsylvania State Department of Health Regulation 433.

monitors on both the air and water effluents of this system, and by monitors to guide the processing of wastes by the evaporator.

The *fourth function* is being accomplished by five trailer-mounted radiation monitoring stations which for approximately two years have been operating at the station site and at other locations in the general area. These stations have been collecting data on the airborne activity of the area, radioactive fallout, and background gamma activity. At the same time, a program of river water sampling, both upstream and downstream of the station, has been conducted. Data have been collected on background activity in nearby vegetation, soil, river mud, and well water supplies. All of this program has continued since station operation began, so that comparative data may be obtained.

9–2.7 Remote viewing. Two closed-circuit television systems monitor conditions inside the plant containers. One utilizes cameras having remote control of focal length and of horizontal and vertical position for general viewing of the areas of the containers which are not accessible during operation. These cameras utilize a single monitor in the control room with means for switching the monitor from one camera to another. They will aid in locating coolant leaks so that such leaks may be isolated by valving off the affected piping. The other closed-circuit television system consists of four independent camera monitor systems for remote viewing of the boiler gage glasses from the main control room.

9–2.8 Operation of control centers. The Shippingport Atomic Power Station is centrally controlled from a single location. After the auxiliary systems have been put into service for normal station operation, the functions of starting up the reactor, bringing it into the power range of operation, starting the turbine, and synchronizing the generator to the electrical power transmission system are performed at the three-section main control console. During operation, all normal control of the station and all information needed to regulate its output, or to take emergency action to limit its output, or to shut it down, are located at this control center. A general view of the control center is shown in Fig. 9–23. Station controls are on a sloped surface console having a short raised portion containing the most vital instruments. Instruments of secondary importance are located on a three-section vertical instrument board set behind the control console. The full console and the full instrument panel form a wide angle "U" so that an operator at any one of the three sections can observe important conditions on the other sections.

The arrangement of controls on the console is graphic in layout. A mimic bus bar has been used to increase the plant imagery. The layout proceeds from left to right to represent the plant from reactor core to

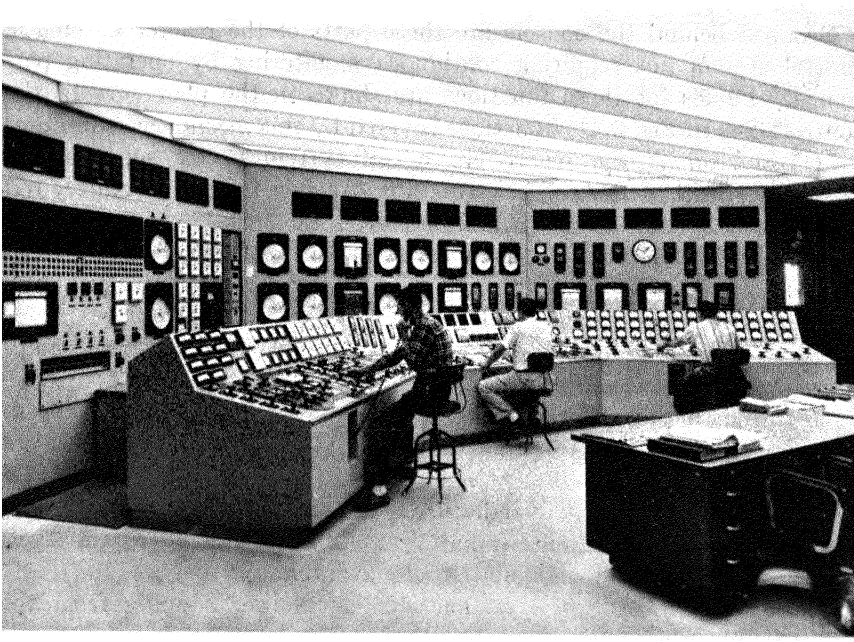


FIG. 9-23. Main control center.

transmission line. During the process of starting up the full station, operations begin at the left, where full monitoring of reactor conditions is concentrated and where control rod motion is regulated. After the reactor has been made critical and has been brought through the subpower range to a low power level, its effects on the reactor coolant loops become apparent. Thus pressurizer representation and reactor coolant loop information are grouped to the right of the reactor on the reactor console. The representation of the heat exchanger is at the junction of the reactor and turbine sections of the console.

Operations for starting up the turbine proceed as soon as the reactor begins to produce significant heat output and the coolant loops are adjusted to proper temperatures. All operations for starting the turbine and monitoring its progress to full speed are directed from the turbine console. When the turbine is at operating speed, the operator at the generator takes over to synchronize the generator to the transmission system. After the station is on the line, normal control of station output is from the turbine section by adjusting the turbine speed and the load limit controls. The reactor operator has no detailed functions to perform if the reactor temperature control is placed in automatic. Shutdown operations are performed in approximately reverse sequence; normal speed manual rod insertion is the last act of the process.

Grouped behind the console are those parts of the reactor automatic control system not requiring continuous monitoring by operating personnel. Panels for these functions are shown in the plan view of the control room in Fig. 9–2. Functions covered by this secondary center are:

(1) Startup of the reactivity control system power supplies and alignment of the normal rod programming sequence from the auxiliary rod control panel. (Control of rod motion is from the console.)

(2) Test of individual rods (with switch-controlled permission from the console) for freedom of motion and determination of torque characteristics from the auxiliary rod control panel.

(3) Energization and checkout of the nuclear instrumentation equipment from the nuclear instrumentation panel.

(4) Setup of pressurizer heater controls for normal operation and checkout of valve operation throughout the plant from the auxiliary control board.

(5) Detailed read-out of radiation levels on the radiation monitoring system from the operational radiation monitoring panel. (Alarm levels are annunciated on the main station annunciator.)

(6) Energization of the station auxiliary power supplies from the 480-volt substation control panels.

(7) Detailed read-out of core temperature and flow instrumentation from the core instrumentation panel.

(8) Energization and checkout of the reactor automatic control system from the power and temperature control panel.

(9) Energization and operation of the failed element detection and location system.

(10) Read-out of miscellaneous plant instrumentation, such as storage tank levels, integrated coolant system flow, etc., from the auxiliary instrument panel.

(11) Control of the station battery charging equipment from the battery charger control panel.

(12) Monitoring of transmission line loading and control of transmission line circuit breakers from the 138-kv instrument panel.

(13) Remote inspection (by closed-circuit television) of the plant container exteriors from the remote viewing system console and of the boiler gage glasses from the boiler level TV monitor panel.

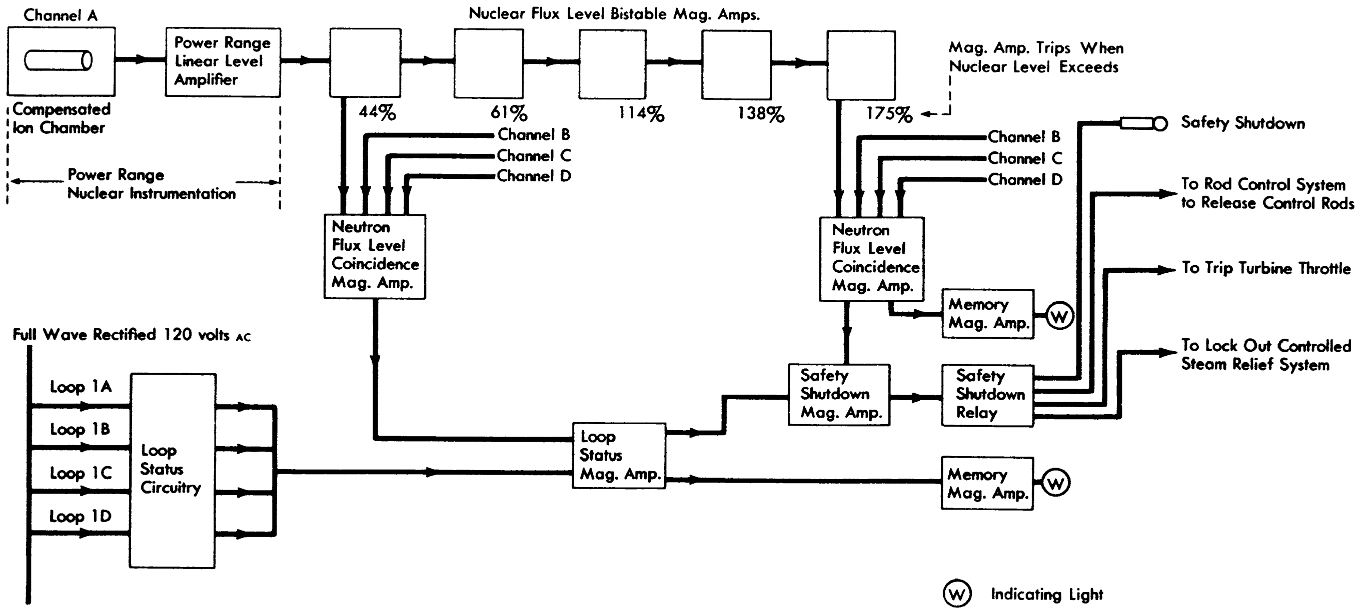


FIG. 9–12. Reactor protection system excessive nuclear level and minimum flow protection safety shutdown circuitry.

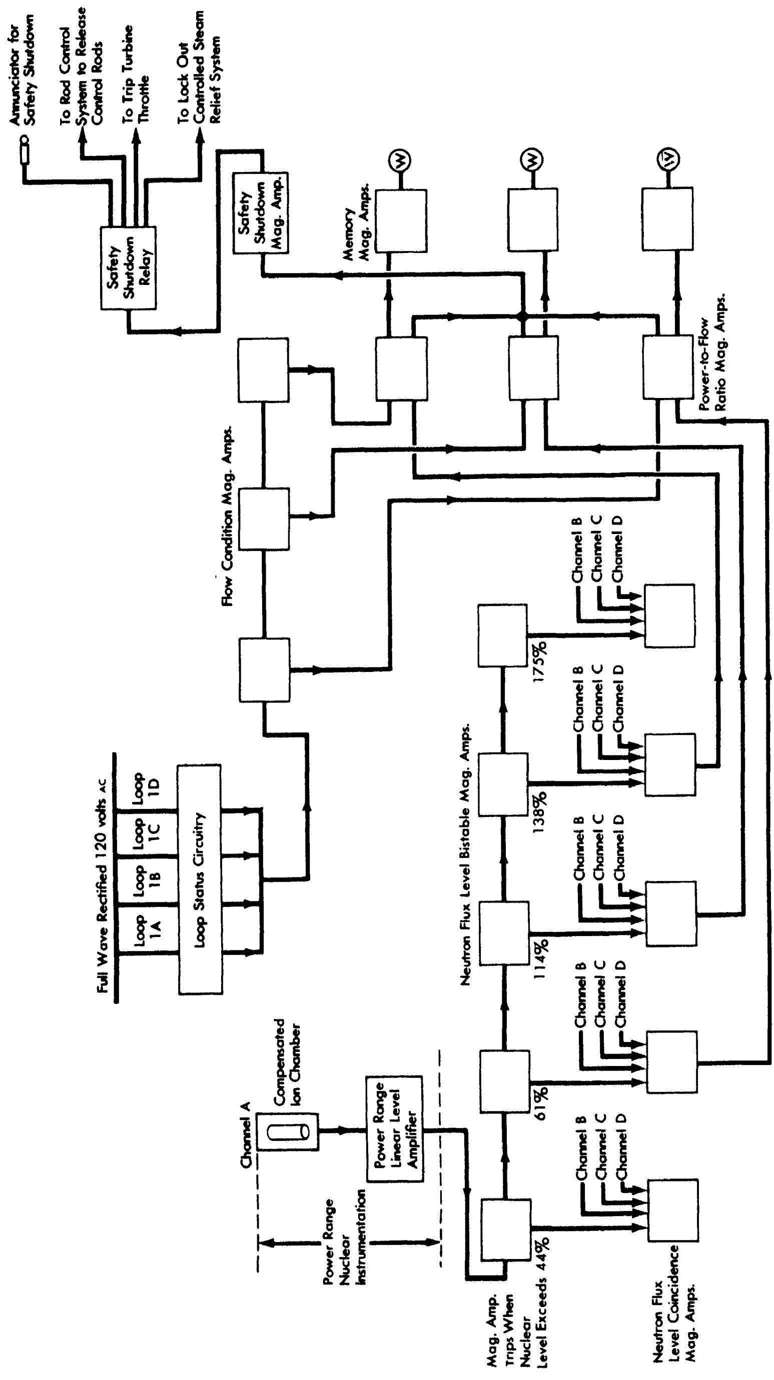


FIG. 9–13. Reactor protection system power-to-flow ratio shutdown circuitry.

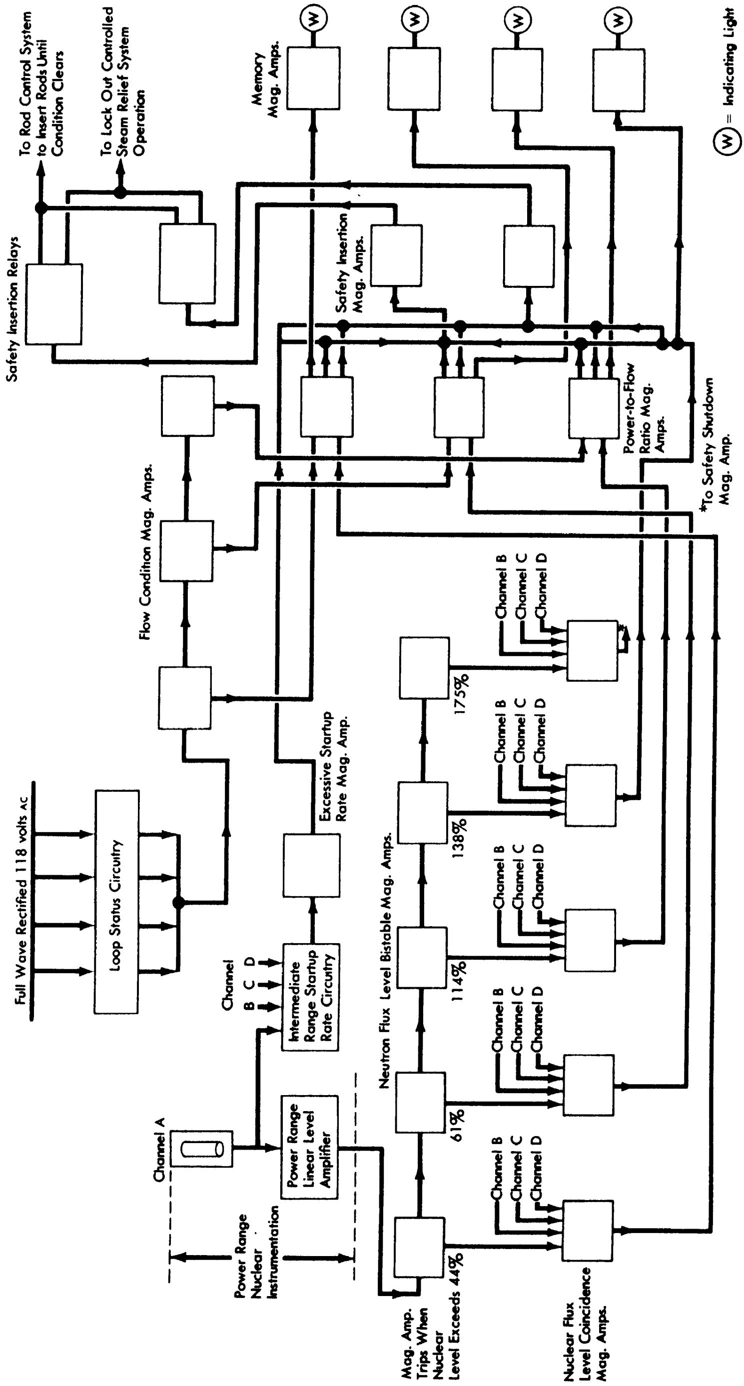


FIG. 9–14. Reactor protection system power-to-flow ratio safety insertion circuitry.

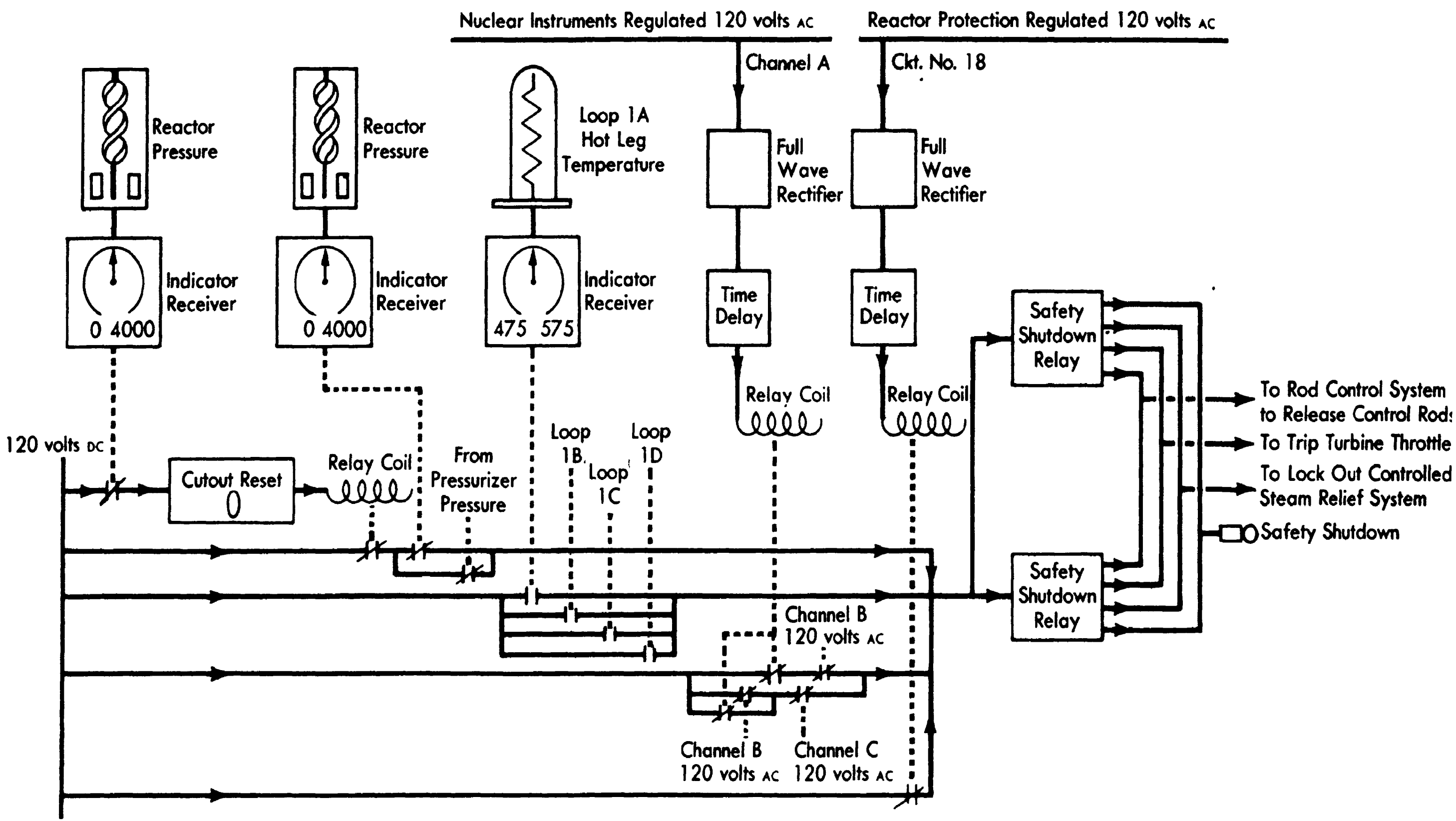


FIG. 9-15. Reactor protection system low coolant pressure, high hot-leg temperature, and undervoltage safety shutdown circuitry.

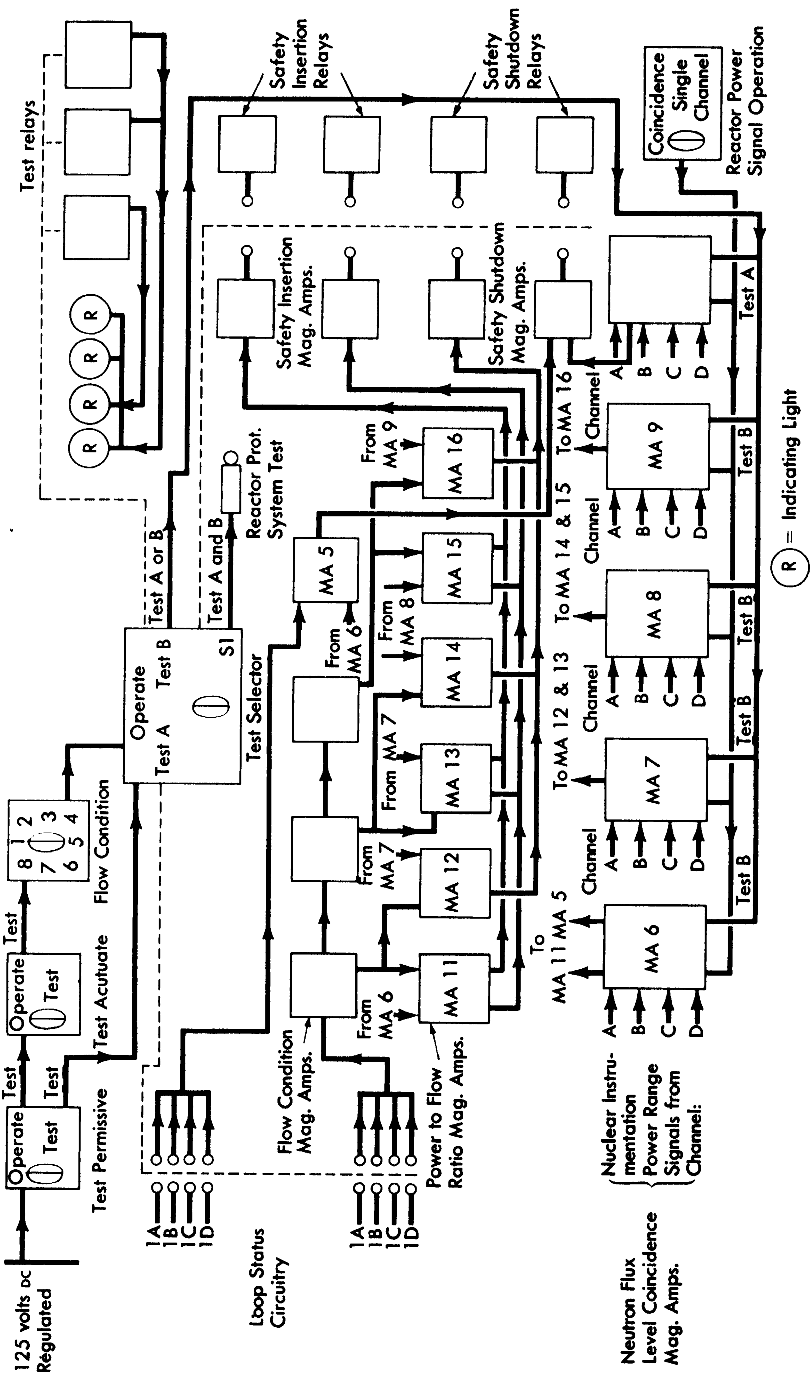


FIG. 9–16. Reactor protection system test set circuitry.

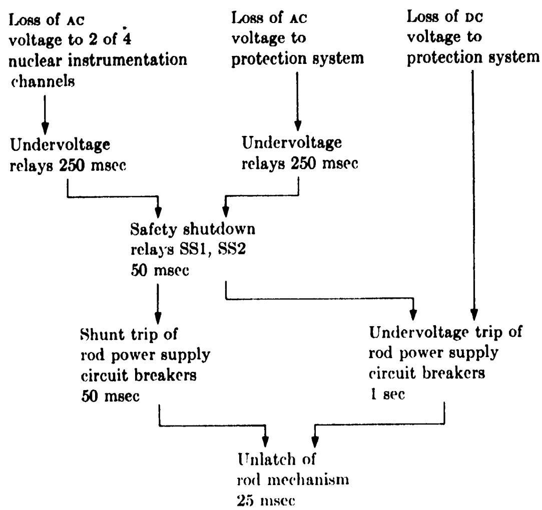


Fig. 9–17. Delay time for shutdown caused by loss of AC and DC control voltage.

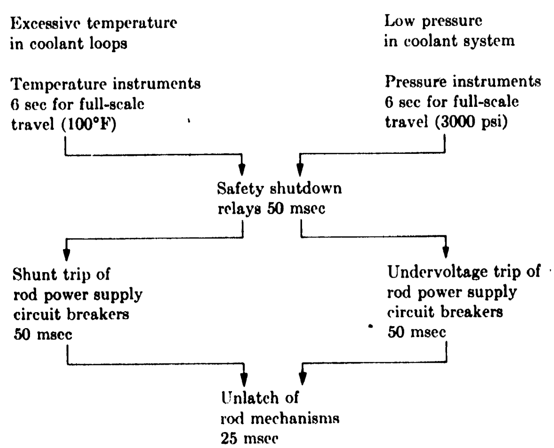


Fig. 9–18. Delay time for shutdown caused by excessive temperature (T_h) and low pressure.

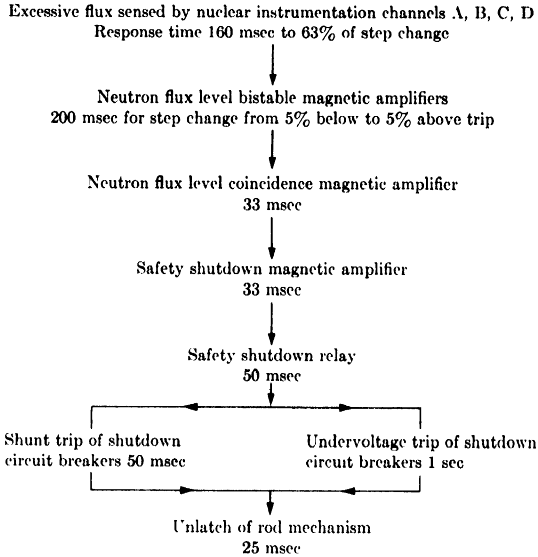


FIG. 9-19. Delay time for shutdown caused by excessive neutron flux at full flow.

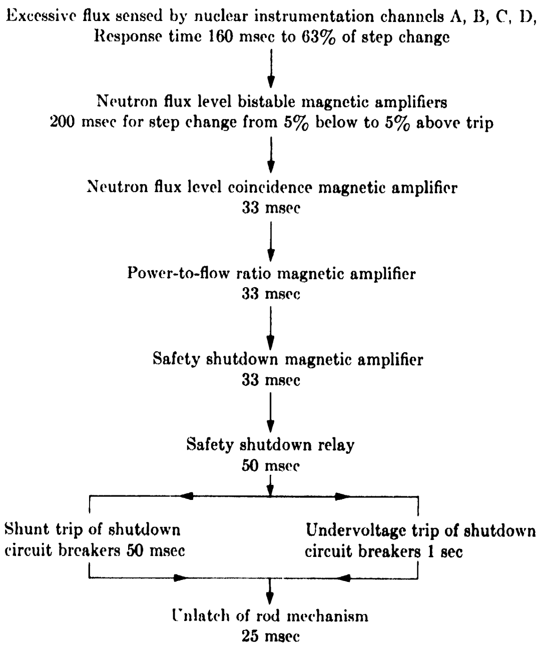


FIG. 9-20. Delay time for shutdown caused by excessive neutron flux at reduced flow.

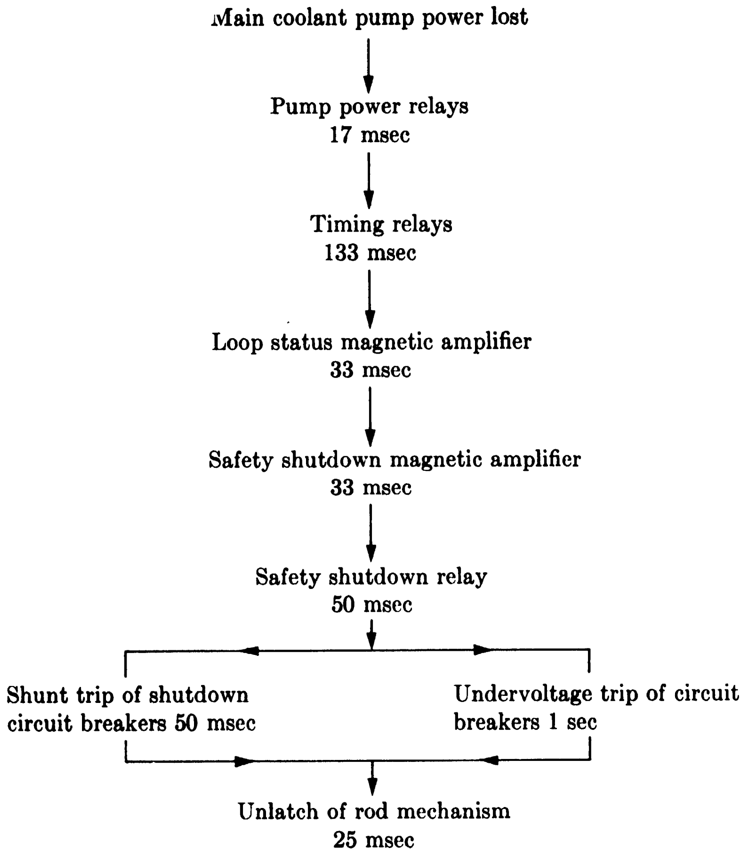


FIG. 9-21. Delay time for shutdown caused by loss of flow (less than 2 loops).

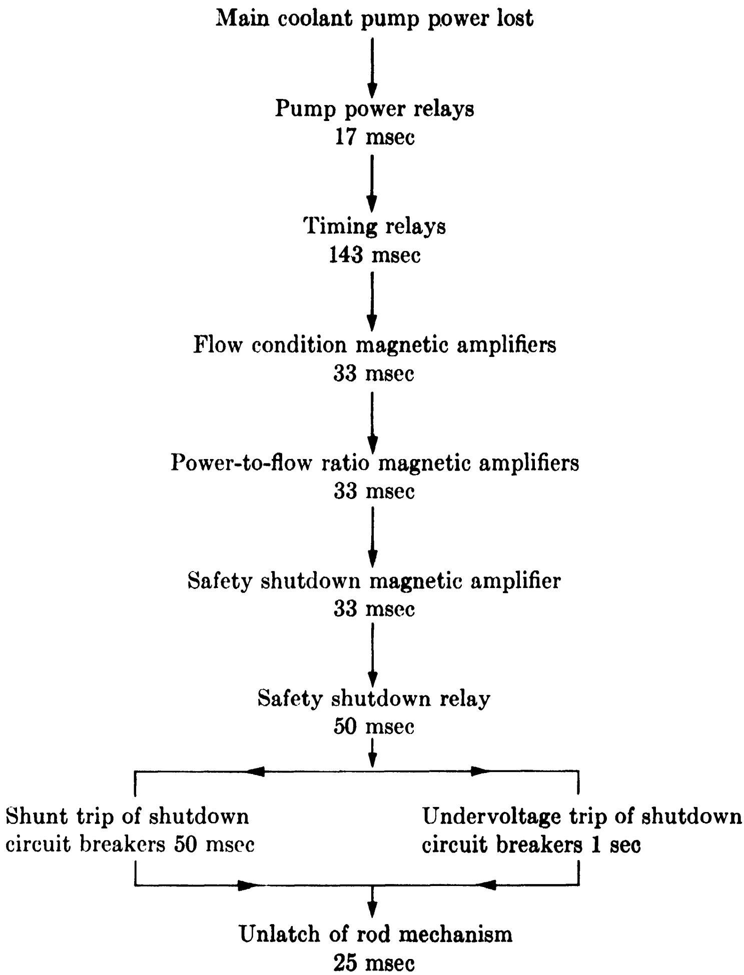


FIG. 9-22. Delay time for shutdown caused by partial loss of flow.

TABLE 9–1

TABULATION OF SYSTEM INPUTS

Plant parameter:	Primary coolant temperature, T_h	Pressurizer pressure, steam	Reactor vessel pressure
Sensing element location	Main coolant loops hot leg between hydraulic and main stop valves	Pressurizer	In pipe to #2 reactor relief valves
Receiver location	Auxiliary instrument panel	Auxiliary instrument panel	Main control console
Type sensing element	Resistance thermometer	Twisted Bourdon tube	Twisted Bourdon tube
Type receiver	Mag amp, servo null balance	Mag amp, servo null balance	Mag amp, servo null balance
Instrument range	475° to 575°	0 to 3000 psig	0 to 4000 psig
Annunciation (A) or insertion (B)	$T_h \geq 544°F$ (A)	<1800 psig (A)	>2150 psig (A)
Shutdown	$T_h \geq 550°$ Any one T_h sufficient	<1600 psig	<1600 psig
Indicator location and type	Meter on aux. inst. panel and recorders on main control console	Main instrument panel	Main control console
Other connections or functions	Power and temperature control	Safety injection and relief valve operation	Safety injection and automatic restoral of low-pressure protection
Comments		Cut out during cold startup until pressure is above 1600 psig	Cut out during cold startup until pressure is above 1600 psig

Plant parameter:	120 volts AC regulated for reactor protection system	120 volts AC regulated for nuclear instrumentation channels	Neutron flux (power range)
Sensing element location	Reactor protection system panel, Section III	Reactor protection system panel, Section I	Detector wells in neutron shield tank
Receiver location			Nuclear instrumentation panel
Type sensing element	Type SG relay	Type SG relay	Compensated ion chamber
Type receiver			Diode in series with mag amp
Instrument range	Drop out, approx. 15 cycles after loss of voltage or reduction ($\frac{1}{2}$ wave)	Drop out, approx. 15 cycles after loss of voltage or reduction ($\frac{1}{2}$ wave)	1 to 175% rated full power
Shutdown	0.25 sec, 2 of 3	0.25 sec, 2 of 3	See *Comments*
Indicator location and type	Lights on reactor protection system panel, Section III	Lights on reactor protection system panel, Section I	Meters on main control console
Other connections or functions	None	None	Power and temperature control
Comments	Relays are supplied by bridge rectifier from voltage regulating transformer	Relays are supplied by bridge rectifiers from AC supplies to nuclear instrumentation channels	Safety insertion and shutdown with associated annunciation dependent on reactor power to reactor coolant flow ratios (2 out of 3 coincidence)

(continued)

TABLE 9-1 (*continued*)

Plant parameter:	Reactor startup rate (source range) not connected	Reactor startup rate (intermediate range)	Primary coolant pump speed
Sensing element location	Detector wells in neutron shield tank	Detector wells in neutron shield tank	Auxiliary power room
Receiver location	Nuclear instrumentation panel	Nuclear instrumentation panel	
Type sensing element	BF_3 counter	Compensated ion chamber	Auxiliary contacts on speed selector switch
Type receiver	Pulse integrator and log microammeter	Log microammeter	
Instrument range	$10^{-9}\%$ to $10^{-4}\%$ of rated full power	1.75×10^{-5} to 175% rated full power	Contacts closed for slow speed connection
Annunciation (A) or insertion (B)	See *Comments* (B)	Startup rate greater than 1.74 decades/Min (B)	See *Comments* (B)
Shutdown			See *Comments*
Indicator location and type	Meters on main control console	Meters on main control console	Lights on main control console
Other connections or functions	Nuclear instrumentation system	Nuclear instrumentation system	Reactor protection system pump power relays and low flow alarm interlocks
Comments	Insertion capability provided same as intermediate range but not connected	2 out of 3 coincidence or any single channel. This protection manually removed above 10% power	Used in determining when pumps are not at full speed

Plant parameter:	Primary coolant flow (pump power)	Primary coolant flow (main loop stop valves)	Primary coolant flow
Sensing element location	2400-volt switchgear and speed changer switch	Main coolant loop stop valves	In each main coolant loop between boiler outlet and pump
Receiver location	Coolant pump relay panel	Auxiliary instrument panel	Main control console
Type sensing element	Potential and current transformers	E-core valve position indicator	Venturi tube and d/p cells
Type receiver	Single phase under power relays	Mag amp and relay	Mag amp, servo null balance
Instrument range	0 to 800 watts balanced 3 phase (120 V PT and 5A, CT)	Open or not fully open	0 to 9 $\times$ 10^6 lb/hr
Annunciation (A) or insertion (B)	Less than 40% pump power at full speed and 75% pump power at $\frac{1}{2}$ speed. See *Comments* (B)	$\frac{1}{2}$-inch valve travel. See *Comments* (B)	75% of rated flow for full speed and 75% of rated flow for half-speed pump operation
Shutdown	Same as insertion. See *Comments*	$\frac{1}{2}$-inch valve travel. See *Comments*	
Indicator location and type	None	Indicator lights on main control console	Main control console
Other connections or functions	Timing relays	Primary coolant pump breakers. Loop temperature interlock circuit	None
Comments	Insertion and shutdown depend on power to flow ratios. Loss of flow is detected when timing relays drop out after pump power relays restore	Insertion and shutdown depend on power to flow ratios. Valve not fully open indicates associated loop not operating	Used for flow indication and annunciation only, not connected to protection system

(continued)

TABLE 9-1 (*continued*)

Plant parameter:	Main generator circuit breaker trip	Manual shutdown
Sensing element location	Generator and transformer leads	Main control console
Receiver location	Generator and transformer relay racks	
Type sensing element	Potential and current transformers	Manually operated switch
Type receiver	Aux. relays operating from differential relays	
Instrument range	See *Comments*	See *Comments*
Annunciation (A) or insertion (B)	For generator shutdown (B)	Safety shutdown
Shutdown		Manually controlled
Indicator location and type	None	
Other connections or functions	Throttle trip and DLCo. generator protection system	Rod control system
Comments	Relays operate for various generator and main and No. 1 station transformer faults	Manual safety shutdown switch may be used to shut down reactor at any time

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CHAPTER 10

RADIOACTIVE WASTE DISPOSAL SYSTEM

CHAPTER 10

RADIOACTIVE WASTE DISPOSAL SYSTEM*

The radioactive waste disposal system (RWDS), shown in Fig. 10–1, is designed to contain and process all radioactive wastes generated by the Shippingport reactor, so that the material discharged does not create a health hazard. Since this is the first radioactive waste disposal system designed for a large-scale power reactor in a populated area, much engineering effort was concentrated on disposal criteria, waste treatment processes, and controls required for satisfactory waste disposal. The major results of this study are presented in this chapter.

10–1. Radioactive Waste Disposal System Requirements

10–1.1 Fundamental disposal requirements. The criteria for disposal of radioactive wastes from the PWR plant are based on National Bureau of Standards Handbook 52, *Maximum Permissible Amounts of Radioisotopes in the Human Body and Maximum Permissible Concentration in Air and Water*, on National Bureau of Standards Handbook 61, *Regulation of Radiation Exposure by Legislative Means*, and on Pennsylvania State Department of Health Regulation 433. These documents recommend maximum permissible concentrations for specific radioactive materials in the environment. To assure that these concentrations are never experienced at the Shippingport Atomic Power Station, the RWDS was designed so that it does not add more than 10 percent of these recommended maximum permissible concentrations to the original radioactivity content of any water discharged to the river and of any gas discharged to the atmosphere.

Liquid criteria. Since data on the quantity and activity of all the specific radioisotopes in the PWR coolant were not available during design of the RWDS, it was decided to design the liquid portion of the waste disposal plant on the basis of the maximum permissible concentration for an unidentified mixture of radioisotopes. Since the maximum permissible concentration for this unidentified mixture is given in the above mentioned documents as 1×10^{-7} μc/ml of water, the concentration in the water discharged from the PWR shall be increased from its original radioactivity content by no more than 10^{-8} μc/ml.

Gaseous criteria. The major radioactive constituents of PWR gaseous wastes have been determined to be 5.3-day Xe^{133} and 10-year Kr^{85}.

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* R. D. Brown, Westinghouse Bettis Plant, and M. Shaw, U. S. Atomic Energy Commission.

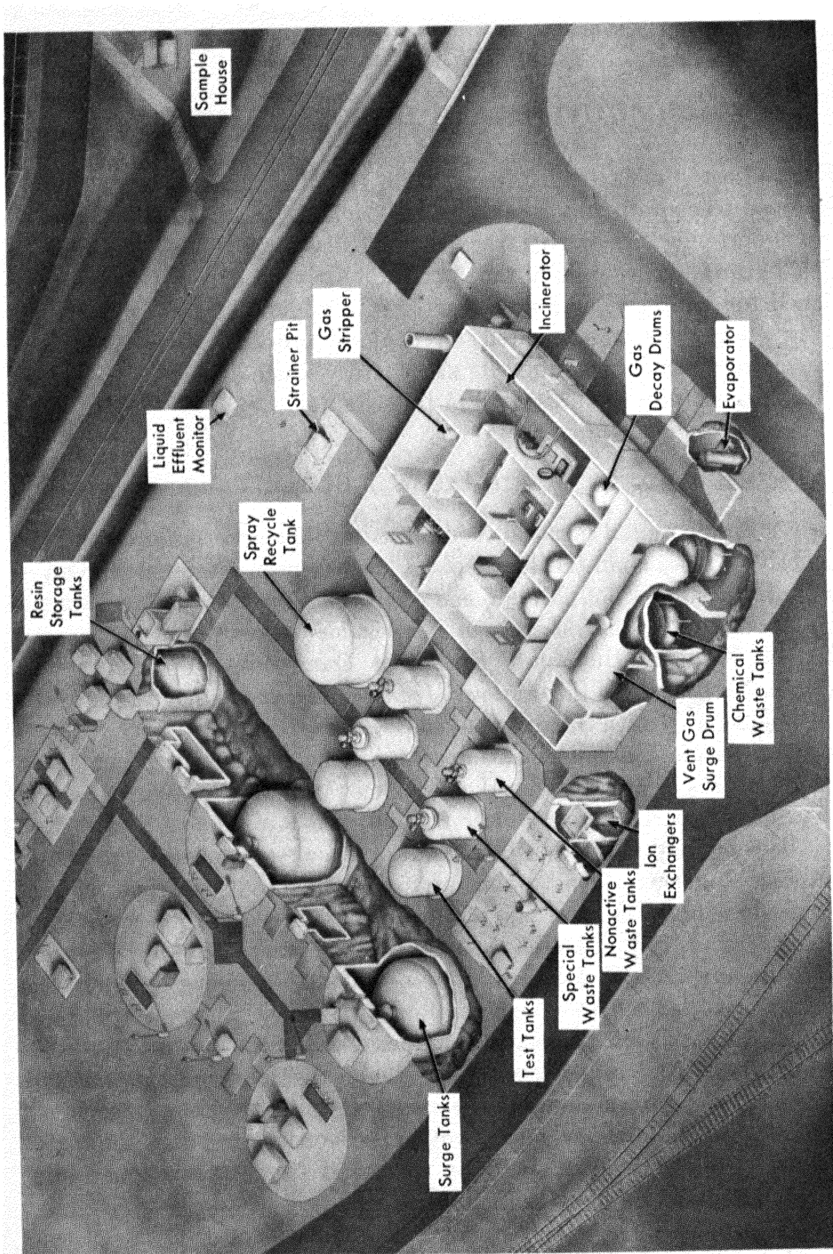

FIG. 10–1. Radioactive waste disposal system.

The maximum permissible concentration of Xe^{133} is shown in NBS Handbook 52 as 4.0×10^{-6} μc/ml of air. The maximum permissible concentration of Kr^{85}, found by utilizing NBS Handbook 52, is 4.0×10^{-6} μc/ml of air. Thus the total activity of the gaseous wastes discharged from the PWR plant shall not be increased by more than 4.0×10^{-7} μc/ml of air.

10-1.2 Additional disposal requirements. In addition to the fundamental disposal requirements stated above, the system was also designed in accordance with the following:

(1) Any tank or burial facility for solid wastes must be protected against leakage into the ground and against vapor release to the atmosphere.

(2) Sufficient capacity must be provided to store radioactive gases for the maximum duration of a weather inversion.

(3) Wastes shipped from the site must be packaged in a manner to prevent a radioactive hazard to personnel involved in the packaging or shipping operations. Regulations for public carriers set forth in the U. S. Interstate Commerce Commission Regulations, Parts 71 to 78, must be observed.

(4) Sufficient protection in the form of shielding must be provided so that the radiation dosages received by operating personnel will be less than the maximum permissible dose rate recommended by the National Radiation Protection Committee.

10-2. Sources and Types of PWR Radioactive Wastes

Radioactive waste from the plant is in solid, liquid, and gaseous forms. PWR wastes can be classified into the following five categories:

10-2.1 Reactor plant effluents. The radioactivity of reactor plant effluents is due to: (1) activation of corrosible metals, corrosion products, and trace elements in the high-purity reactor coolant water; (2) fission products released from failed fuel elements; and (3) tritium resulting from the reaction that occurs when the lithium added to the reactor coolant via the coolant purification system absorbs neutrons.

Activated corrosion products. High-purity water is charged to the plant, and the materials in contact with the reactor coolant are highly corrosion-resistant stainless steel. Nevertheless, some corrosion of the surfaces in contact with the reactor coolant will occur, even though suitable corrosion inhibitors are injected into the coolant. This amount of corrosion would be considered insignificant in a conventional power plant, but in a reactor power plant minute quantities of impurities become activated in passing through the reactor core and constitute one source of

radioactivity in the reactor plant effluents. These effluents are discharged to the RWDS by the discharge and vent system (see Chapter 8).

Fission products. The source of the fission products is the reactor fuel itself. The PWR reactor utilizes enriched uranium and unenriched uranium dioxide (UO_2). as fuel. The largest portion of the fuel is UO_2. The enriched uranium is encapsulated in high-integrity containers which are not expected to rupture or release radioactive materials to the reactor coolant. The UO_2 is contained in zirconium alloy tubes with welded ends. These tubes are not considered to have as high an integrity as the containers for the enriched uranium. For the design of the waste disposal system it was assumed (1) that one percent (1000) of the UO_2 tubes might develop pinholes during the life of the core, and thus (2) that at some time near the end of core life, the core might contain 1000 failed fuel elements that could release radioactive fission products to the coolant.

Tritium. The reactor coolant water is treated with lithium resins to minimize corrosion and deposits in the system. Lithium was selected in preference to other alkalis because of its good nuclear properties and the effectiveness of the lithium-hydroxyl mixed bed ion exchanger in removing high-hazard radioactive fission products from the coolant stream. The lithium atom reacts with neutrons to form an alpha particle and a tritium atom. Tritium (12-year half-life) decays very little during any reasonable holdup time. As a part of water, it cannot be removed or concentrated by distillation, chemical treatment, or ion exchange. Fortunately, tritium is a relatively low-hazard isotope, compared with other isotopes, which accounts for the fact that the tolerance level allowed by the Commonwealth of Pennsylvania and recommended in NBS documents is comparatively high. The RWDS contains the tritiated water and releases it at a controlled rate. The tritiated water occurs only in the relatively low volume (750 gallons per day) reactor plant effluent.

Radioactivity in the reactor coolant is collected and concentrated by the coolant purification system, which supplements the chemical treatment given the coolant. To limit radioactivity buildup, a fraction of the coolant that passes through the reactor core is purified by circulation through a bypass demineralizer. The demineralizer removes insoluble matter and replaces soluble matter with lithium hydroxide. Because of the nature of the demineralizer resin, lithium present in the coolant is not removed. Tritium, being chemically combined in the water molecules, is also unaffected. When the resin is depleted, it is transported to resin storage tanks in the RWDS.

10–2.2 Service building wastes. Waste from the service building consists of the effluents from the laundries, showers, laboratories, and the

special cleaning facilities used to decontaminate mechanical equipment prior to performing maintenance operations. Since the radioactivity level of these effluents varies considerably, it is desirable to segregate them into three classes based on the estimated radioactivity levels.

"A" wastes, also known as "monitored wastes," are from laundering noncontaminated clothing, and from showers, lavatories, drinking fountains, and monitored laboratory sinks. They have little or no radioactivity.

"B" wastes, also known as "special monitored wastes," consist primarily of wastes from the laundering of suspected contaminated clothing and from special monitored laboratory drains. The activity of the B wastes is expected to be greater than that of the A wastes.

"C" wastes, also known as "decontamination room wastes," are from the decontamination cleaning baths. These wastes contain a greater amount of radioactivity than either A or B wastes and also may contain chemicals that will be neutralized before processing in the waste disposal plant.

10–2.3 Combustible solid wastes. These wastes include rags, paper, and wooden material that may have become radioactively contaminated.

10–2.4 Noncombustible solid wastes. These consist of spent ion exchange resins, incinerator ash resulting from the burning of radioactive combustible materials, and contaminated products such as tools and equipment. The incinerator ash and spent resin are transported through pipes in the form of an aqueous slurry.

10–2.5 Radioactive gaseous wastes. These wastes are radioactive gases, primarily xenon and krypton, formed in the reactor plant by the fission of uranium.

10–3. Groundwork Prior to Selection of PWR Disposal System

10–3.1 Volume and radioactivity level of wastes. To initiate the design of the RWDS, the quantity and radioactivity of the various wastes to be processed had to be determined. Prediction of the over-all quantity of wastes was based on the expected frequency of maintenance operations and on experience gained from handling radioactive products in other installations. The radioactivity levels of the various wastes were predicted in several ways. The activity of the reactor plant effluents (both liquid and gaseous) was predicted by in-pile tests and subsequent calculations on fuel elements containing simulated defects (pinhole leaks). These fuel elements were similar to those in the PWR. Predicted activity levels of the monitored and special monitored wastes were based on the experi-

ence of other installations. The activity levels of the other wastes were conservatively estimated, since no applicable experience was available.

10–3.2 Design criteria. No fuel reprocessing facilities were included in the design of the PWR plant. The problems associated with PWR waste disposal are, therefore, considerably less than those at a plant which includes radiochemical processing. It is to be emphasized, however, that because of the developmental nature of the Shippingport plant with regard to future cores and possible destructive testing of core subassemblies, considerable latitude was used in choosing both the methods of disposal and the storage capacity of the waste disposal system. Future "nondevelopmental" pressurized water nuclear plants are expected to require considerably less storage capacity and less versatile but simpler waste disposal facilities.

In determining which of the many possible disposal methods would be suitable for PWR, the following factors had to be evaluated:

(1) Whether the process had been commercially proven or developed, at least to pilot plant scale.

(2) Whether the process would reduce the activity level of the wastes enough to permit safe disposal of the processed fluids to the environment without exceeding specified tolerance levels.

(3) Whether the process would be too costly.

(4) Whether the proper controls would be available to protect operating personnel and people in the surrounding area.

10–3.3 Fundamental methods considered. Several methods of disposing of the various types of radioactive waste material are employed at the National Laboratories and other AEC installations. Many others have been suggested and proposed in AEC documents. There are four general types:

(1) Natural decay.

(2) Dilution and dispersion.

(3) Land or sea burial.

(4) Concentration and storage.

A study was undertaken to determine which methods were best suited to the specific requirements of the PWR plant. The principal results of this study follow.

10–4. Selection of a Disposal System

10–4.1 Liquid waste disposal. As a result of research at the Bettis Plant, some of which is described below, ion exchange was chosen as the principal

method of high-purity liquid waste disposal. In addition, limited vapor compression evaporation is used for processing wastes with high chemical content. Radioactive decay and dilution are utilized as far as practical. The way in which the problem of liquid waste disposal was solved is outlined in the following paragraphs.

Ion exchange. When the design program for the PWR radioactive waste disposal system started, experimental data were lacking to determine decontamination factors which could be achieved with processed wastes consisting of pure, deionized water containing carrier-free radioisotopes. Consequently, a research program was begun at the Bettis Plant to evaluate ion exchange as a possible method of radioactive liquid waste disposal. Results of in-pile tests of PWR-type fuel elements, containing simulated (pinhole) defects, indicated that the principal contributors to gross nonvolatile water activity from fission products in the PWR wastes are isotopes of iodine, cesium, rubidium, barium, strontium, and molybdenum. Others are zirconium, niobium, ruthenium, tellurium, and the rare-earth elements.

Experiments at Bettis Plant demonstrated that decontamination factors at least as great as required by the PWR waste disposal system design are obtained during ion exchange processing of high-resistivity solutions of carrier-free isotopes, representative of alkali metal, alkaline earth, rare earth, ruthenium, rhodium, zirconium, and niobium fission products in the PWR reactor coolant. In addition, to determine the rate of depletion of ion exchange resin, Bettis Plant studied high-resistivity water containing the above-mentioned representative isotopes at levels equivalent to the expected maximum steady-state levels of the corresponding fission products in the PWR coolant. It was found that large volumes of this water could be processed by ion exchange (with reasonable resin consumption) before activity break-through.

Evaporation. Liquid wastes of high chemical content cannot be processed through the ion exchange system, as described above, without sacrificing resin or resin bed performance. Hence, a limited vapor compression evaporation process was chosen for this processing.

Evaporation concentration and subsequent land or ocean burial of the concentrate has been employed satisfactorily on a large scale by various AEC installations. The techniques, operating procedures, and decontamination factors for the evaporation process are well known from extensive operating experience.

Practical, proven designs have been developed for evaporators in which distillate activity is less than one millionth that of the bottoms liquor (decontamination factor 10^6). Essentially this has been achieved by designing highly efficient deep-bed vapor filters to remove entrained liquid particles. Close pH adjustment of wastes (slightly alkaline) has

adequately prevented the volatilization of iodine and ruthenium in the feed solution.

Despite its merits, evaporation is relatively costly. Evaporation costs are estimated at from 10 to 13 cents per gallon of feed liquid. In addition, a study to determine the feasibility of evaporating all PWR radioactive wastes revealed that the process was too complicated. Thus evaporation for disposal of all liquid wastes was ruled out.

Decay. Should 1000 fuel elements fail, an over-all reduction of approximately 2×10^8 in activity level would be required for the nonvolatile fission products in the PWR wastes before discharge to the environment. It was apparent that this factor could not be practically achieved by natural decay processes alone, since the gross decay rates for the nonvolatile fission products in the PWR reactor effluents were calculated to be as follows:

Decay interval	*Gross decay factor*
Shutdown to 1 hour	4
1 hour to 24 hours	5
1 day to 3 days	1.2
1 day to 11 days	2.2
1 day to 16 days	3.2
1 day to 31 days	10
1 day to 46 days	31
1 day to 61 days	73
1 day to 76 days	99

Direct dilution. By proper blending with enough uncontaminated water, radioactive liquid wastes may be disposed of without exceeding permissible tolerance levels. If an adequate supply of uncontaminated water is available, this is perhaps the least expensive method, and requires minimum handling.

Although the Shippingport plant is advantageously situated on the Ohio River, it was decided early in the design stages not to use this large river for direct dilution of liquid wastes. This decision was influenced by the remote chance that inadequate mixing of the wastes and river water might contaminate areas downstream from the Station. Instead, it was decided to dilute the wastes to design concentration (see Article 10–1.1) with only that quantity of river water diverted into the plant for cooling the main condenser. By controlling the discharge of low-level wastes properly, the limited dilution process utilized at the PWR is closely supervised by Station operating personnel who are provided with suitable alarms and instruments to protect against accidental discharge. Furthermore,

the additional dilution provided by the full river flow serves as an added design safety factor.

Disposal by ocean burial. The possibility that all radioactive aqueous wastes could be buried in the ocean has been explored. The procedure would be modified from practices of other AEC installations to suit the special conditions of the PWR. Waste liquor received from the reactor plant would be mixed with dry cement in 55-gallon steel drums and the mixture allowed to solidify. The drums would then be trucked to a marine terminal, loaded aboard ship, and buried at sea.

Ocean burial without concentration was found too expensive because of the large quantities of radioactive liquid wastes anticipated at the PWR. A cost estimate indicated a range from $3.50 to $10.00 per gallon, with as much as 30,000 gallons for disposal each month.

Montmorillonite clay. It is well known that certain clays exhibit a cationic exchange capacity. The efficiency of this process depends on the type of clay used and the chemical composition of the waste solution. Brookhaven National Laboratory has pioneered the possibilities of using montmorillonite (a nonswelling bentonite clay) for this purpose on the basis of its superior ion exchange and mechanical stability when extruded into columns. An attractive method is the adsorption of radioactive ions from an aqueous solution onto montmorillonite clay, followed by a high-temperature sintering (1000°C) which locks the adsorbed material into a nonleachable state. This method allows simple disposal of the sintered product in the ocean or on land without danger of disseminating the contained radioactivity.

This process has been successfully demonstrated. However, at the time the radioactive waste disposal facilities of the Shippingport plant were initially designed, it was concluded that the process would require much developmental and pilot plant work before it could become routine commercial procedure.

10–4.2 Solid waste disposal. Combustible solid wastes are incinerated. The resulting gases are disposed of as described in the article on gaseous waste disposal. Noncombustible solid wastes, including incinerator ashes and spent demineralizer resin, are stored on the site. This method was selected because it appeared simplest and safest, and involved the lowest initial and operating costs. Other noncombustible solids will be disposed of by ocean burial or by underground burial at the site. The way in which the problem of solid waste disposal was solved is outlined in the following paragraphs.

Storage. The spent resin is flushed from the reactor plant and RWDS demineralizers and discharged in the form of an aqueous slurry directly into underground storage tanks, without use of special handling equip-

ment. After the resin settles, the liquid is removed for further processing.

Filtration. One disposal method considered was flushing the spent resin out of the reactor plant demineralizers into a filter, then removing it from the filter and conveying it in bags or containers to a concrete vault. This was discarded because of disadvantages which include the following:

(1) It would have required the development of remote handling techniques and the use of expensive remote handling equipment.

(2) A tanklike, watertight steel liner inside the concrete vault would have been needed to prevent seepage of radioactive water from the resin into the ground.

(3) Some indirect means of cooling the solid material to remove the decay heat would have been required.

Incineration. Burning the resin in an incinerator was also considered. Because intermediate storage is needed, this method of disposal has all the disadvantages of the filtration method. Its only advantage is that a smaller volume of solid goes to final storage. In addition, the incinerator would require heavy shielding to protect operating personnel. Probably most important, the radioactive gases from the incinerator would require special washing and treatment before discharge to the atmosphere. Consequently, this method was not considered applicable to PWR, except for the combustible wastes.

10–4.3 Gaseous waste disposal. The nature of the waste gases from the Shippingport plant led to the selection of a decay and dilution disposal method. Reasoning leading to this choice is outlined below.

Decay and dilution. Because the fission gases from the PWR plant are principally 5.3-day Xe-133 and, to a lesser extent, 10-year Kr-85, decay and dilution were found practical for their disposal, on the design basis of 1000 failed fuel elements. By allowing the waste gases to decay for 60 days, their activity could be reduced by a factor of approximately 2500. Coupled with a reasonable quantity of air dilution, this would reduce the gaseous activity to allowable tolerance at discharge from the plant.

Dilution and dispersion. The standard procedure at AEC installations for disposal of fission product gases released during the processing of spent fuel elements is dilution with air and discharge to the atmosphere from a high stack. Disposal of waste fission product gases from PWR by this method alone would require a volume of air so large that it would be uneconomical.

Liquefaction. Liquefaction, another disposal method considered practical, was still in the pilot plant stage at the time the PWR system was designed. This method consists of liquefying the fission gases and requires expensive low-temperature equipment.

10-5. Description of Waste Disposal Process

Summary flow sheets depicting the waste disposal processes are shown as Figs. 10-2 and 10-3, and process layout sheets are shown as Figs. 10-4 and 10-5.

10-5.1 Reactor plant effluents. The waste effluent (reactor coolant) from the reactor plant may contain highly radioactive soluble and insoluble materials as well as dissolved radioactive gases. The estimated average total volume of these wastes is 3160 cubic feet per month.

Reactor plant effluents are treated by: (1) decay, followed by (2) ion exchange, and finally (3) separation (followed by decay) of the soluble gases from the liquid. The gross activity of these wastes is due to both volatile and nonvolatile radioactive materials. The gaseous or volatile activity of the waste is estimated to be a maximum (with 1000 failed fuel elements) of 8.7 $\mu c/ml$ of liquid, while the nonvolatile activity is estimated to be a maximum of 1.4 $\mu c/ml$. After treatment, the radioactivity of the waste is estimated to be a maximum of 4.5×10^{-5} $\mu c/ml$ for the non-

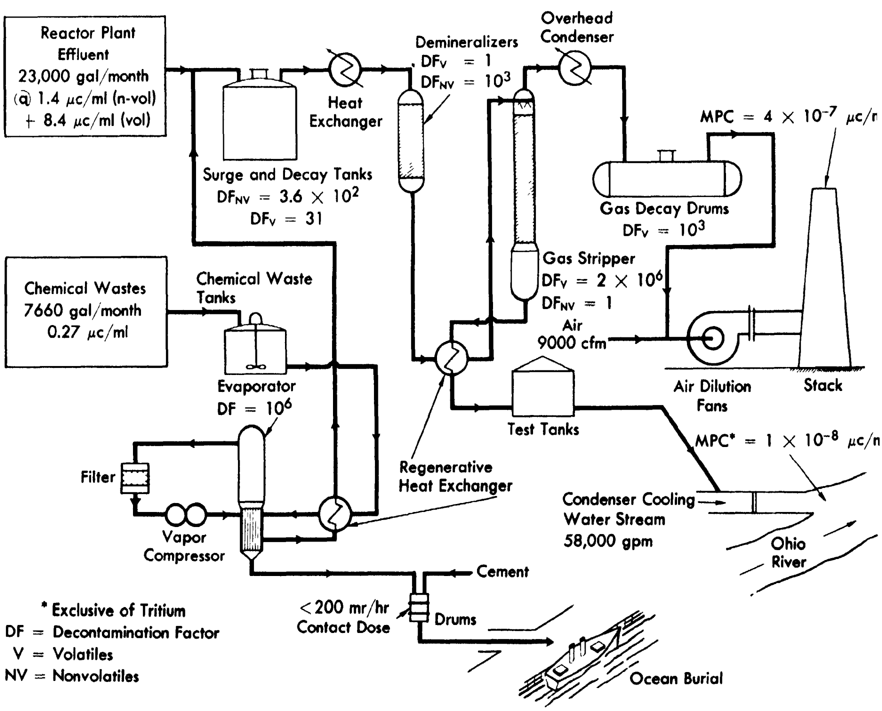

Fig. 10-2. Radioactive waste disposal facilities for liquids and gases ("A" and "B" wastes not shown).

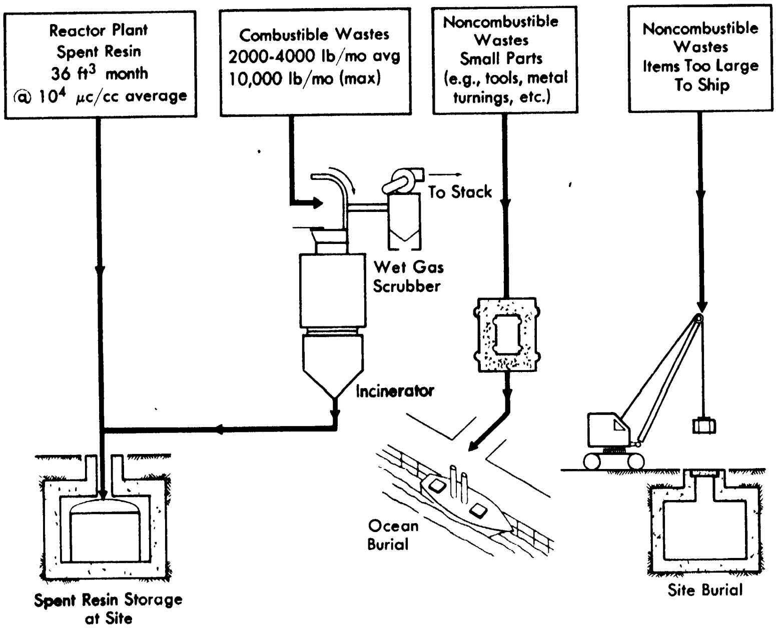

FIG. 10–3. Radioactive waste disposal facilities for solids.

volatiles and 1.2×10^{-8} μc/ml for the volatiles. The waste is monitored in the system's test tanks and the proper flow rate determined so that after blending with the turbine condenser cooling water stream the activity on leaving the plant will be no more than 1×10^{-8} μc/ml.

To ensure the desired degree of treatment, the liquid waste is sampled and radioactivity determined at each stage of the process. The entire treatment is a series of batch processes, and the waste may be reprocessed if samples indicate that radioactivity has not been adequately reduced.

The reactor plant effluents are first transported to stainless steel surge and decay tanks which provide storage volume for these effluents for as long as 45 days. During this interval, radioactivity will decrease by an estimated factor of 31 for nonvolatiles and an estimated factor of 360 for volatiles. The surge and decay tanks are completely below ground, to protect personnel against radiation.

Waste liquid is sampled in the decay tanks to ensure adequate reduction of radioactivity before further processing. Next, this liquid is treated in ion exchangers to reduce the radioactivity further. Nonvolatile radioactivity is decreased by a factor of at least 1000 in these ion exchangers; the volatile activity is not appreciably reduced.

Liquid from the surge and decay tanks is first passed through a heat exchanger which cools the liquid so that its heat will not deplete the

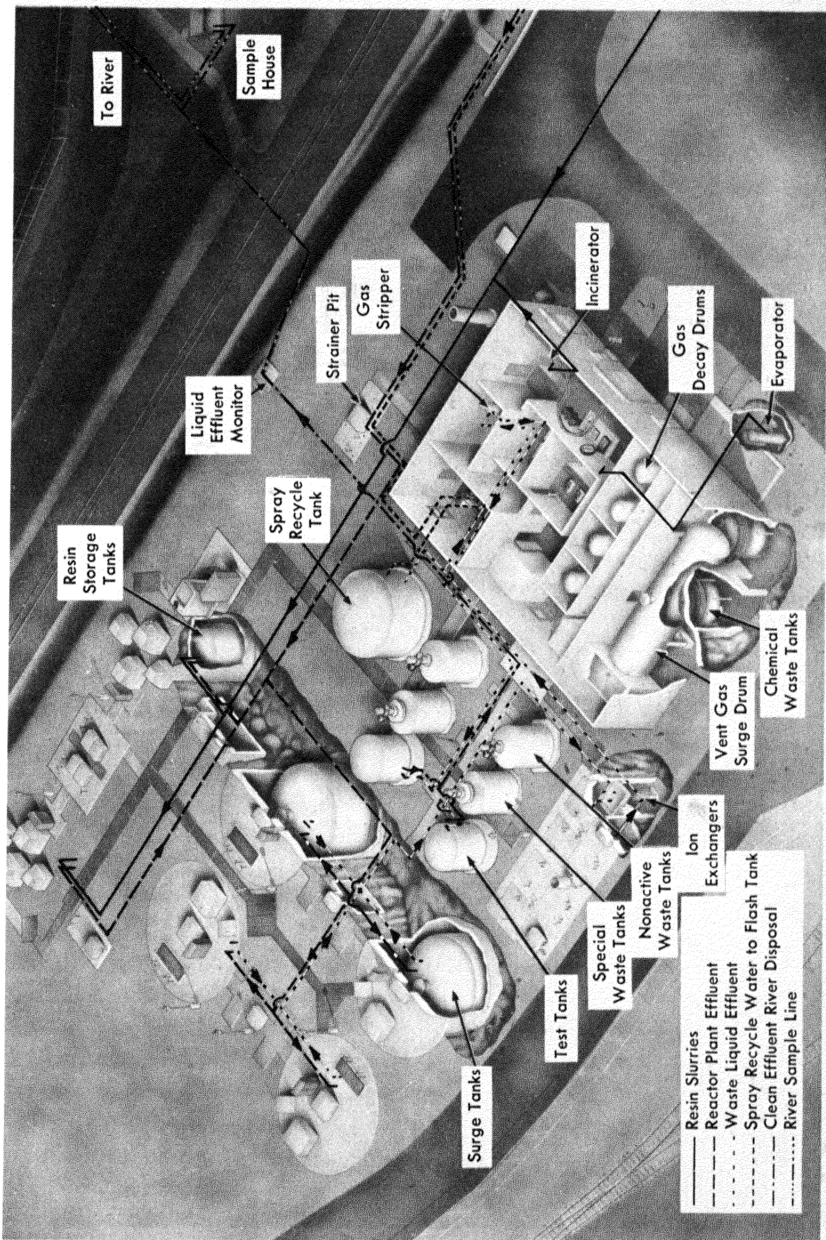

FIG. 10–4. Radioactive waste disposal facilities—main process stream.

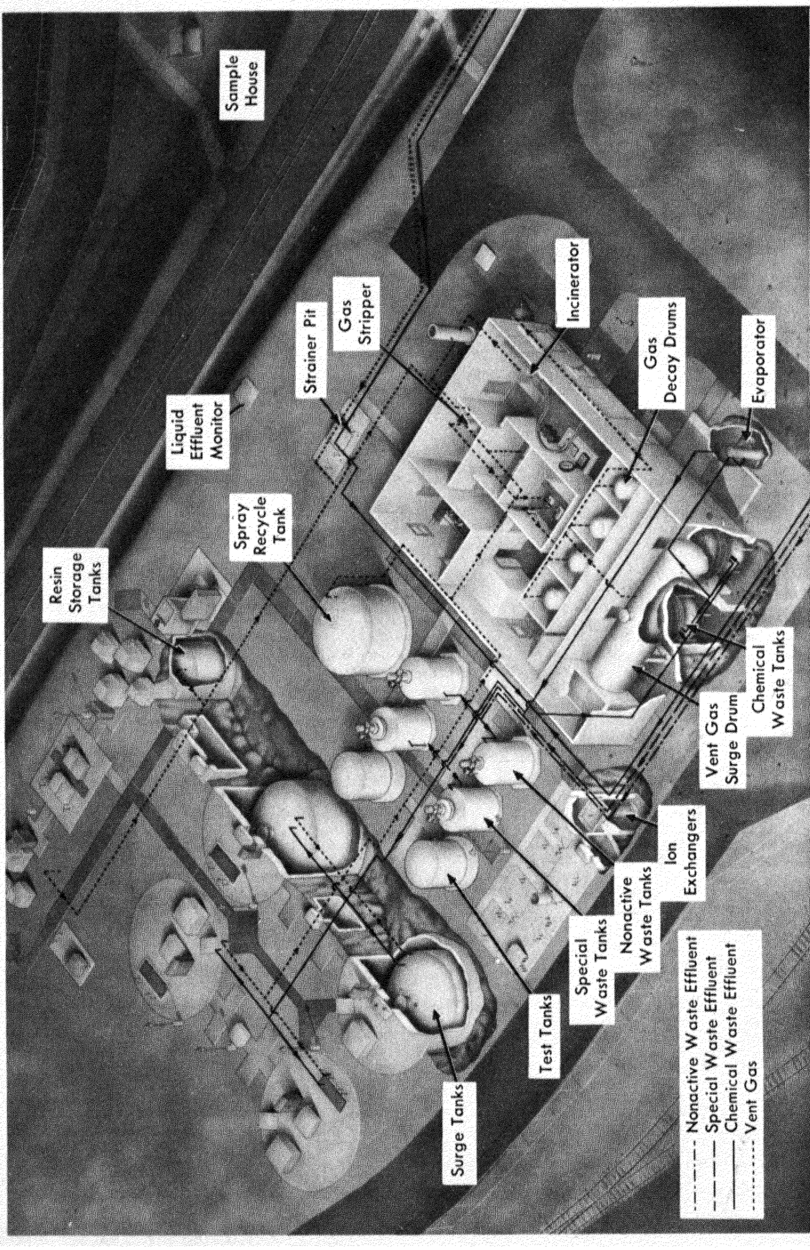

FIG. 10–5. Radioactive waste disposal facilities—secondary process stream.

capacity of the ion exchange resin. The liquid then flows through a series of four ion exchangers which are manifolded so that their relative positions can be altered. To shield personnel, the ion exchangers (demineralizers) are enclosed by concrete below ground level in the waste disposal area.

The liquid effluent from the ion exchangers may be diverted to the spray recycle tank. This tank stores the cooled, deionized reactor plant effluents and recycles them as necessary to cool the contents of the coolant discharge and vent system flash and blowoff tanks (see Chapter 8).

The waste liquid not diverted to the spray recycle tank is sent to a gas stripper, where most of the radioactive gases are removed. Activity due to volatile content of the liquid is reduced by a factor of 2×10^6 in the stripper, while the nonvolatile activity is not significantly affected.

Liquid waste from the stripper is then piped into one of two available test tanks at ground level. Each has an operating capacity of 5000 gallons, and is fabricated of carbon steel and lined with a plastic material. Shielding is not required, since the radioactivity is very low. The tanks hold the processed liquid until analyses determine whether the desired reduction in activity has been obtained. Normally, one of the test tanks is being filled while the other is being emptied (after satisfactory analysis) to the turbine condenser cooling water discharge stream. The waste liquid is pumped to this stream through a continuously recording flow regulator and activity monitor.

The waste liquid is discharged into the condenser cooling water through a distributor located in the condenser cooling water effluent channel. The blended liquids then pass over a weir, where further blending occurs. In the effluent channel downstream of the weir is a sampling rake which obtains samples from 21 different locations in the channel. These samples are mixed and a representative sample collected and analyzed to determine whether the activity of the liquid leaving the plant conforms to the established RWDS criteria.

10–5.2 Service building wastes. *Monitored wastes* are the effluents from the showers and "cold" (nonradioactive) laundry. This effluent is not radioactive under normal conditions but is processed as a safety precaution. The monitored wastes are pumped from the service building into the "A" tanks in the waste disposal area. The estimated volume of this waste stream is 22,600 ft^3/month and the estimated (maximum) radioactivity (all due to nonvolatiles) is 8.1×10^{-7} μc/ml. Liquid from these tanks is monitored for radioactivity in the laboratory and then discharged to the effluent channel, where it is diluted with the turbine condenser effluent stream prior to discharge to the Ohio River.

Special monitored wastes are the effluents from the "hot" laundry and laboratory drains. The activity of this stream may be considerably

greater than that of the monitored wastes. The special monitored wastes are pumped from the service building into the "B" tanks located in the waste disposal area. Estimated volume of this waste stream is 11,850 ft^3/month and the estimated (maximum) activity (all nonvolatile) is 2.2×10^{-4} μc/ml. The liquid from these tanks is monitored for radioactivity in the laboratory, then pumped to the effluent channel at a controlled flow rate to produce a blended activity of 10^{-8} μc/ml or less. If the activity should be so great that the tank cannot be emptied in the normal operating time, the liquid can be transferred to the decontamination room waste tanks for evaporation.

Decontamination room wastes are the effluents from the decontamination or cleaning rooms (wastes from decontamination of equipment, tools, etc.). These also are pumped from the service building into the "C" tanks in the waste disposal building. The estimated volume of this waste stream is 1025 ft^3/month and the estimated maximum activity is 0.27 μc/ml. The method of treating this stream, which may contain a high level of dissolved solids, is to concentrate the solids, mix the concentrate with cement in standard 55-gallon drums, and bury them at sea. The decontamination room waste stream is neutralized with caustic soda or sodium carbonate in the "C" tanks and fed into a vapor compression evaporator. The evaporated liquid (overhead) is condensed and sent to the surge and decay tanks where, as a safety precaution, it is processed as part of the reactor plant effluents. The concentrated liquor in the evaporator is drummed and buried at sea. The stainless steel evaporator unit, which has a capacity of 100 gal/hr, is of the standard vapor compression type and is located in a shielded vault beside the waste disposal building. The vault is not normally occupied, since all evaporator operations are remote.

10–5.3 Scheduling of liquid wastes discharged from plant. The RWDS was designed so that the assumed maximum daily accumulation of each of the three types of liquid wastes (i.e., from the surge and decay, monitored, and special monitored tanks) may be individually discharged from the plant in less than 24 hours should the activity of all these streams go as high as the design criteria. The design activity values are believed to be conservative and, therefore, the corresponding operating activity values will be lower. The time required to discharge the wastes individually from the plant will be lessened in proportion to the lower activity values.

10–5.4 Gaseous wastes. Radioactive gases, predominantly xenon and krypton, are formed in the reactor and dissolved in the water which is drained as part of the reactor plant effluent. This gas enters the waste

disposal system with the liquid wastes. Most of this gas remains in the liquid during the entire waste treatment process until the liquid enters the gas stripper. The steam stripper, operating under a slight vacuum, reduces the gas remaining in the liquid to an insignificant amount and passes the stripped gas to the waste disposal vent gas system.

In the vent gas system the gas from the stripper is compressed and piped to a large surge tank. At infrequent intervals the gas from the surge tank is discharged to one of four gas decay drums. The gas in the decay drums is stored long enough (60 days design based on 1000 failed fuel elements) for the radioactivity level to decrease until, by controlled discharge and dilution with 9000 ft^3/minute of air from standard industrial fans, it meets the tolerance standards.

The waste disposal vent gas system includes a "breathing system" for the resin storage tanks, the surge and decay tanks, the spray recycle tank, and the discharge and vent system flash tank. The system, completely enclosed, receives and stores all gases from these tanks, and maintains a constant pressure in the vapor spaces of the tanks as they "breathe" with changes in liquid level.

The major equipment in the gaseous waste system consists of the vent gas surge drum, four gas decay drums, three gas compressors, the gas stripper, and two air dilution fans. All this equipment is in the waste disposal building. Shielding is provided by eight-inch thick concrete walls, which separate the equipment from personnel passageways in the waste disposal building.

10–5.5 Solid wastes. *Combustible solid wastes* are transported to the waste disposal building, where they are incinerated. The flue gases are sent through a wet gas scrubber in which particulate matter five microns and larger is removed, and the gases are cooled. Scrubber effluent liquid may be discharged to the surge and decay tanks or to the resin storage tanks. The cooled flue gas is then passed through an exhaust filter for final cleanup before discharge to the stack. Residue ash from the incinerator is dumped into the ash slurry tank below the incinerator. After being slurried with water, the ash is pumped into the resin storage tank. An estimated 2000 to 4000 lb of combustible waste will be processed each month. The activity of this waste may vary over an extremely wide range.

Noncombustible solid wastes consist of resins from the plant's ion exchangers, residue ash from the incinerator, solids from strainers in the pipelines and from the incinerator stack exhaust filter, and contaminated items of plant equipment. The method of disposing of this last category is to embed the equipment in concrete and dispose of it by ocean burial if the item is relatively small. If a large piece of equipment is involved, it is embedded in concrete and buried at the site.

Resins, residue ash, and strainer and filter solids are slurried with water and pumped into one of two resin storage tanks. After the solids have settled, the water is piped into one of the surge and decay tanks. This liquid is then processed in the same manner and with the same equipment as the reactor plant effluents.

The resin storage tanks are in the waste disposal area in individual, totally enclosed vaults several feet below ground level. Shielding is provided by the concrete enclosure and the earth cover.

The incinerator and the ash slurry tank are in the waste disposal building. These stainless steel units require no special shielding because they are remotely operated and because the equipment is in a pit in the building.

10–6. Control of Waste Disposal Process

10–6.1 System monitoring. As mentioned above in the description of the waste treatment processes, the waste streams are sampled as they enter the RWDS, before and after each waste treatment process, and before discharge from the system. There are also sampling facilities that monitor liquid and gaseous waste streams after dilution. The system is flexibile enough that any waste stream may be reprocessed if necessary to reduce the radioactivity to a level from which subsequent dilution will yield a stream at or below the maximum permissible concentration. This flexibility, along with the basic batch type of processing in the system and the large number of sample locations, will minimize the possibility of discharge of over-tolerance wastes from the plant.

In addition to the numerous sampling provisions, effluent activity monitors are installed on the main liquid effluent header and on the stack. These monitors will detect and inform the operator if either liquid or gaseous wastes over tolerance are accidentally discharged.

10–6.2 Environment monitoring. As mentioned in Chapter 9, a monitoring program was started early in 1956 and is continuing during operation of the plant. The preoperational phase was to determine the types and amounts of radioactive materials occurring in the Shippingport plant environment and the variations in the amounts of these materials during the two years or so before startup of the nuclear portion of the plant. This should permit good evaluation of environmental radioactivity, so that the studies being conducted while the plant is in operation will reveal significant changes due to plant operations. Evaluations have been and are being made on (1) soil in the general vicinity of the plant, (2) Ohio River water above and below the site, (3) well water within a one-mile radius of the site, and (4) vegetation and (5) air in the general area.

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CHAPTER 11

HAZARDS EVALUATION

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HAZARDS EVALUATION*

11–1. Approach to Hazards Evaluation

Prevention of off-site radiation hazards was one of the prime considerations in the design of the Shippingport plant and will continue to be a prime consideration in all training, operation, and testing at the Station. This consideration is over and above the normal precautions taken for the safety of personnel and equipment.

The approach was to make the containment barriers for radioactive material as reliable as possible and then to evaluate those accidents which could result in a release of any radioactive material from these barriers. Results of these evaluations were incorporated in the plant design as it progressed.

This chapter contains a summary assessment of the potential hazards of plant operation, an outline of the steps taken to make the containment barriers reliable, an evaluation of each of the types of accidents which might lead to release of radioactivity and of the steps that have been and will be taken with respect to operation and testing to prevent such accidents, and a hazards evaluation of fuel handling and radioactive waste disposal.

The results of the hazards evaluation have been published in a series of reports (the first fifteen listed at the end of this chapter), which also embody the experience gained from the design and operation of other pressurized water reactor plants. The reports are available from the Office of Technical Services, U. S. Department of Commerce, Washington 25, D. C.

11–2. Summary and Assessment of Accident Conditions

The Shippingport reactor will not present a hazard beyond the site boundary as the result of an explosion, either physical, chemical, or nuclear in origin. In addition, the plant has been designed so that no feasible sequence of events could cause a release of hazardous quantities of fission products beyond the site boundary.

Inherent in the plant design are two independent, continuously monitored barriers between the fission products and the surrounding area.

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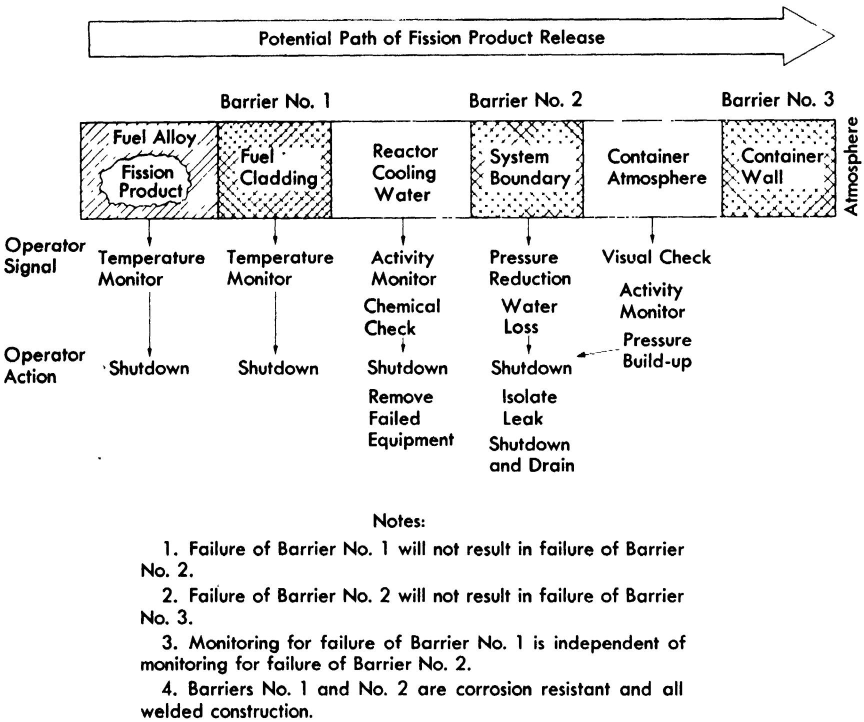

FIG. 11–1. The three independent barriers to fission product release.

First, the fission products are contained within the fuel elements by the cladding, a corrosion-resistant boundary which is designed, fabricated, and inspected in a way that results in a high degree of integrity.

The fuel elements are cooled by reactor coolant water circulating past them in an all-welded, pressurized system. This second barrier for containment of fission products is also very reliable, continuously monitored, and completely independent of the integrity of the fuel elements. If some fission products leak into the water through flaws which may develop in the fuel element cladding, these fission products will be contained within the reactor coolant system. Provisions have been made to locate and remove faulty fuel elements if the coolant radioactivity should become too high.

In addition to these two inherent barriers and as a third independent barrier, the entire reactor plant is housed in a group of interconnected vapor-tight steel pressure vessels called the reactor plant container.

These three independent barriers are shown schematically in Fig. 1–1. The path that fission products would have to take to be released to the atmosphere outside the container is shown in Fig. 11–1. This figure also indicates the signals available to the plant operators at each step in the

path and the action that they would take on the basis of these signals to prevent release of fission products.

All accidents which were considered pertinent to the safety of the plant have been analyzed. These included loss of station power, loss of coolant flow to the core, reactivity accidents, loss of reactor coolant through a rupture in the reactor coolant system, and accidental release from the radioactive waste disposal system. Conclusions of these studies may be summarized as follows:

(1) The design of the plant is such that there would be no hazard to the personnel in the plant or to the surrounding area in case of complete loss of electrical power.

(2) The reactor protection system is designed to prevent any damage to the core, and no combination of pump failures or other loss of coolant flow will result in a release of fission products from the fuel elements.

(3) The inherent stability and design characteristics of the reactor system prevent the occurrence of any nuclear transient that would result in vaporization or gross melting of the fuel elements with an attendant large pressure buildup in the reactor coolant system. As a consequence, no casualty is envisioned whereby a nuclear excursion could result in a rupture in the reactor coolant system with a release of significant quantities of fission products from the fuel elements. This conclusion includes consideration of the possibility of a chemical reaction of zirconium with water.

(4) The release of 100% of the reactor coolant to the outside atmosphere would not result in a biological hazard at the site boundary even if the core had been operated for 3000 hr at a power level of 275 Mw and the coolant contained the maximum expected activity, i.e., the activity caused by imperfections in the cladding of about one percent of the UO_2 fuel elements (1000 defected elements during the lifetime of the core).

(5) No significant radiation hazard would result from the maximum possible accidental release of gaseous or liquid wastes from the radioactive waste disposal system.

(6) Complete loss of reactor coolant through an assumed major rupture in the reactor coolant system is the only accident which could cause a release of any significant amount of fission products to the plant container. This is very improbable, but if such an accident should occur, the safety injection system is capable of pumping sufficient quantities of water into the reactor coolant system to prevent core melting and consequent development of a biological hazard beyond the site boundary. This applies to all locations and sizes of rupture except the one described below.

If a rupture larger than 4 to 6 in. in diameter occurred in the portion of the reactor coolant system pressure boundary below the level of the core and near the reactor vessel, there could be some core meltdown, some

zirconium-water reaction, and some release of fission products to the plant container. Based on the minimum expected flow of 1500 gpm from the safety injection system, a maximum of 20% of the blanket region could melt. However, it is estimated that the biologically significant fission products which would escape from the core to the container would be only about 1.4% of the total activity in curies at shutdown. This estimate is based on the major portion of the fission products remaining contained within the fuel material during meltdown and only selected fission products being released.

During such an accident a maximum of 4.5% of the 9.38 tons of zirconium associated with the heat transfer areas in the core could react with steam or water. The zirconium-water reaction could not start for some minutes after the rupture and would be in the nature of a rapid and progressive oxidation, neither explosive nor in the nature of a self-propagating fire.

The reactor plant container can adequately contain the maximum pressure developed as a result of this worst-case coolant system boundary rupture, which would release all of the reactor coolant and the secondary water from one boiler. The heat from a possible subsequent exothermic zirconium-water reaction and from combustion of the hydrogen product as it is evolved would result in only a minor increase in pressure within the plant container, at a time when the total pressure would be considerably less than the container design pressure. If the hydrogen gas does not burn as it is evolved but mixes uniformly with the air and steam within the plant container, the hydrogen concentration will be below the flammable limit and neither burning nor detonation should occur.

It is calculated that the total external radiation dose that would be accumulated as a result of seepage from the plant container after such an accident, measured at a point on the site boundary 1700 ft from the plant container, would be about 1.0 r. If a person were at this point for a period of 12 hr, the integrated dose to the thyroid gland resulting from inhalation of the iodine fission products would be about 500 rep. The integrated three year dose to the bone resulting from inhalation of strontium fission products would be about 0.20 rep. The Atomic Energy Commission, in WASH-740, "Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants" (March 1957), has proposed an acceptable emergency dose to be one from which persons are assumed to have received no injury and is taken to be 25 r of whole-body gamma radiation in one brief exposure or 50 r in 3 months. This reference also sets 2000 rep to the thyroid gland and a lifetime dose to the bones of 50 rep as reasonable acceptable emergency doses. It is concluded, therefore, that the worst-case loss-of-coolant accident presents no biological hazard to the uncontrolled area surrounding the site.

It is concluded that the plant and reactor have been so designed and constructed as to reduce to a minimum the probability of release of significant quantities of radioactivity in the event of the worst feasible accident. The plant is housed inside a sealed container as an extra precaution against the release of activity. The decision to use a container was made before the design of the plant was started. It was, therefore, not built as a result of any analysis that showed it to be necessary. In fact, there is doubt as to the need for the container.

11-3. Containment of Radioactivity

The basic plant design of the PWR provides two containment barriers between the fissionable fuel and associated core fission products and the surrounding area. These are the fuel element cladding and the piping and vessel walls of the systems containing reactor coolant. A third containment barrier, the reactor plant container, completely surrounds the reactor plant and is independent of the other two containment barriers. These three barriers are discussed in detail in the following sections of this chapter. The fuel handling and waste disposal facilities are external to the plant container and are considered from a hazards viewpoint near the end of this chapter.

11-3.1 First containment barrier—fuel element cladding. The Shippingport core contains two types of fuel elements: seed plates and blanket rods. These are described in detail in Chapters 4 and 6.

Cladding integrity of seed fuel plates. The seed fuel plate type of element has undergone extensive in-pile operation without exhibiting any indications of cladding failure or related accidents. To insure integrity, the PWR plates were each inspected radiographically, metallurgically, and dimensionally according to exacting standards and were tested and inspected for surface flaws and corrosion resistance before being installed in the core. Finished plates were tested for three days in 750°F steam for corrosion resistance. The plates were also tested for accurate location of the fuel alloy within the cladding. The fuel alloy itself is relatively corrosion resistant and metallurgically compatible with the cladding. While no imperfections of the cladding are anticipated, if some should occur, only minor corrosion of the exposed fuel alloy is expected, because of the integrity of the alloy. This minor corrosion would produce no mechanical deformation of the plates and only negligible release of radioactive material to the coolant.

Cladding integrity of blanket fuel rods. Assurance of cladding integrity of the blanket fuel elements was attained by analytical investigations, developmental tests, and production tests.

The analytical investigations were aimed at calculating operational stresses in the cladding that might result from differential temperatures, differential expansion between the fuel and the cladding, and thermal distortion of the fuel bundles, as well as the stresses that would be imposed by system pressure, hydraulic thrusts, and latching spring loads. The results of these calculations were used to ensure a safe design of the fuel element and to establish dimensional tolerances on the grouping of fuel rods in a definite lattice.

The developmental tests (Chapter 5) included both in-pile and out-of-pile tests to provide data concerning the performance of sample fuel elements under irradiation, in contact with high temperature water, and with postulated types of mechanical cladding failure such as buckling of a rod caused by axial loads and yielding or collapse of the cladding caused by external pressure. The results of all these tests indicated that the cladding design is conservative.

The production tests ensured that all material used met the rigid specifications established for physical properties and chemical analysis, including allowable impurity content. Also, the tubing was given a 100% inspection for wall thickness by an ultrasonic resonance method to ensure a minimum thickness of 0.021 in., as established by analytical studies, and a 100% inspection was made for minor flaws by an eddy current test. All flaws discovered were disposed of by cutting out the tubing section containing the flaw. All fuel rods were corrosion tested in degassed water at 750°F for 3 days. After inspection, the fuel rods were welded into fuel bundles and corrosion tested in degassed water at 680°F for 3 days. Welding was performed in accordance with proven techniques and a 100% inspection of all welds was made by radiograph and by helium leak test (Chapter 6).

Long term corrosion tests have demonstrated that UO_2 fuel pellets exposed to high temperature water are stable for long periods of time. Means are provided to detect and locate failures, and to remove faulty fuel elements. The plant has been designed to operate with the fission products in the reactor coolant system that would result from a number of defected rods if cladding failures should occur.

11–3.2 Second containment barrier—reactor coolant system. In addition to the barrier for the containment of radioactivity provided by the fuel element cladding, a vital second barrier to the escape of radioactivity is the pressure boundary of the systems containing reactor coolant at operating temperature and pressure (the reactor coolant systems). The carbon steel vessels in the reactor coolant system have been designed, fabricated, and installed in accordance with exacting specifications to minimize the possibility of a failure in these vessels. All applicable state

and ASME codes have been met. Furthermore, the systems will be pressurized only at temperatures where brittle fracture does not occur. Only ductile materials were used for the reactor coolant and reactor coolant auxiliary piping, since experience with such materials indicates that failures such as small cracks are much less likely to progress into major ruptures than with materials of reduced ductility. With ductile materials, there may be sufficient time after the penetration or crack appears to detect the leak and take remedial measures.

While it is not possible to guarantee that a sizable leak will never develop in the reactor coolant systems during the entire life of the plant, an extensive effort has been made to prevent any failures and it appears that the occurrence of a major break is improbable. In particular, a sudden brittle fracture of any part of the reactor coolant systems is not considered to be feasible.

11-3.3 Third containment barrier—reactor plant container. Should both the other barriers fail, the spread of radioactivity is restricted by completely enclosing the reactor plant in four interconnected steel pressure vessels, called the reactor plant container (Chapter 16). These vessels are in concrete compartments partially below ground level. The container is of sufficient size to contain the expanding steam from release of all the hot pressurized reactor coolant plus the secondary water from one steam generator after a large reactor coolant system rupture, should it ever occur. The design pressure is 52.8 psig and the container was tested pneumatically at 70 psig. A leak test, performed after plant completion and just prior to initial criticality, showed that the container and all of its penetrations were tight.

The possibility of a missile (an object moving with high velocity) of internal origin damaging the plant container and permitting the release of its contents to the outside atmosphere was considered. In order for a failure of the reactor coolant system to result in the ejection of such a missile, it is believed that a brittle fracture would have to occur. It has been noted, however, that the reactor coolant systems are not subject to brittle fracture since they are made of ductile, austenitic, stainless steel and thus are not expected to be a source of missiles. The rigorous material specifications and rigid inspection, fabrication, and field erection procedures for the reactor plant were designed to eliminate possible sources of failure. As additional assurance against a potential missile, all piping, components, and instrument penetrations in the reactor coolant systems were rechecked to ensure that no pressure-containing parts are overstressed. It is concluded that no feasible missile, capable of puncturing the plant container, could be ejected simultaneously with a rupture of the reactor coolant system.

11–4. Loss-of-Reactor-Coolant Accident

11–4.1 Definition and possibility. A loss-of-coolant accident is defined as one in which coolant water escapes as a result of a rupture of a vessel or pipe in the reactor coolant system pressure boundary. The reactor coolant system is continuously monitored, and if any leak exists, the plant can be shut down.

Every precaution has been taken through rigid material and fabrication specifications, inspection procedures, stress analyses, testing, and control of operating temperatures to ensure that all equipment is made of high quality material, correctly designed, properly manufactured, and correctly installed. It is believed, therefore, that the probability of rupture has been reduced to a minimum. However, a major reactor coolant system pipe or vessel rupture, although extremely unlikely, is a potential hazard to the surrounding area. Therefore, the loss-of-coolant accident was investigated in detail.

An investigation was also made of the possibility of another accident (a loss of coolant flow or a reactivity excursion) causing an overstressing of the pressure boundary of the reactor coolant system and resulting in a loss-of-coolant accident. It was concluded that there are no feasible accidents which would result in a pressure rise so high that the reactor coolant system would be overpressurized. For example, the reactor relief valves can discharge coolant at a rate that will prevent pressure rising above 2500 psi, even if the turbine throttle should trip when the plant is producing 100 Mw of electrical power and the reactor should fail to scram.

11–4.2 Sequence of events during accident. The sequence of events that would occur following a rupture of the reactor coolant system pressure boundary depends on the size and location of the rupture. In case of a serious rupture, the water level in the pressurizer and the reactor plant pressure would begin to fall and a low-water level alarm would sound. Shortly thereafter, when the pressure dropped to 1850 psia, another alarm would automatically sound in the control room. If the pressure should continue to fall, a scram would occur at 1600 psia. At this point, the operator would initiate a prescribed emergency procedure which calls for, among other things, depressing a button on the control console which causes the safety injection system to start automatically when the pressure falls to 500 psia.

The safety injection system is described in Chapter 8. It utilizes the boiler feed pumps to pump water into the reactor pressure vessel to cool the core in the event of a rupture in the reactor coolant system. A study was made of the possible interaction between a loss-of-coolant accident and loss of power to the station, from which it was concluded that the loss-of-coolant accident could not cause a loss of power to the station or

vice versa. There appears to be no reason to assume that both accidents would occur simultaneously from two unrelated causes. Therefore, it is assumed that electrical power would be available to the safety injection system during a loss-of-coolant accident and that the system would function as designed.

Regardless of the size of the rupture, if it were located above the level of the core, the safety injection system would cover the core with water in a short time, preventing any significant melting of the core or release of fission products. If the rupture were large and if it were located below the core, the safety injection system would take considerably longer to cover the core, because the reactor chamber of the plant container would also have to be filled.

11-5. Loss-of-Coolant-Flow Accident

11-5.1 Definition and significance. A loss-of-coolant-flow accident occurs when the coolant water flow through the reactor is accidentally stopped with the reactor at power or is accidentally reduced below that required to adequately remove the heat being generated by the reactor. Reactor coolant flow can be reduced or lost in one of two ways—the main coolant pumps can be stopped or the remotely operated stop valves can be closed.

The sequence of events involved with such an accident would be as follows: As the reactor coolant flow started to decrease, the reactor would initially continue the production of heat at about the same rate as before the accident. Thus both the water and fuel element temperatures would increase as the flow decreased. As the reactor coolant temperature increased, the negative temperature coefficient of reactivity would act to reduce power. Therefore, if flow were reduced slowly so that the negative temperature coefficient could accomplish a corresponding reduction in power, the accident would merely result in a reduction in the power output of the reactor. If, however, the flow decreased at a rate so fast that the negative temperature coefficient could not reduce power at a corresponding rate, then the excess heat being generated would result in a temperature rise of the fuel elements. If the reactor were not scrammed at this point, boiling would take place on the surface of the fuel elements. If the boiling condition persisted for a short time, the fuel element might become hot enough to melt, since steam is not as effective a heat transfer medium as water. However, this melting would be self-limiting and would result in local burnouts only.

In order to prevent the occurrence of this full sequence of events, which might do costly damage to the core, the reactor protection system would scram the reactor. It is not considered feasible that this accident

would progress to the point of melting any part of the core. However, even if this should occur, the only result would be damage to the core, with some contamination of the internal surfaces of the reactor plant piping and components. There would be no release of radioactivity to the plant container and therefore none to the surrounding area. It is also important to recognize that, since the reactor is completely flooded with water during this accident, any melting of zirconium that might occur would not result in an autocatalytic chemical reaction and subsequent major meltdown of the reactor.

11–5.2 Closing of all stop valves. The main coolant stop valves are installed for maintenance purposes only and there is no normal operating procedure that calls for the closing of *any* of the main coolant valves when the core is producing significant quantities of heat. There is no procedure that calls for closing *all* the main coolant stop valves under any circumstance. This could happen only as a very unlikely error by the operator, since there would be a number of indications and alarms. The reactor protection system would scram the reactor and there would be at least 45 mins in which to take corrective action, which consists merely of turning a switch to open a pair of main coolant valves so that the decay heat removal system could operate to prevent core damage. The possibility of all stop valves remaining closed is, therefore, not considered to present a significant hazard.

11–5.3 Stoppage of main coolant pumps. In considering ways that reactor coolant flow can be lost by the stopping of the main coolant pumps, several types of accidents must be examined. Consideration must be given to the mechanical or electrical failure of a pump and motor as well as to the failure of the source of power feeding a pump motor. Consideration must also be given to the possibility of these failures causing the simultaneous stopping of all pumps as well as to the possibility of pumps stopping in sequence with the failures occurring singly or in groups, spaced by various intervals of time. The worst case turns out to be a particular sequential loss of pump action.

It is not considered reasonable to assume that two or more pumps would fail simultaneously, or nearly so, as a result of trouble within the pumps. The power supply circuitry is so arranged that a single system or equipment fault, or failure of a single source of power, will result in the loss of no more than two pumps. Thus it is extremely unlikely that power would be lost to three or four pumps simultaneously. It is equally unlikely that a power supply failure which causes the loss of two pumps will be followed within a matter of minutes by a second system failure, causing the loss of power to the pumps remaining in operation. However, even should this occur, the core is adequately protected by the reactor protec-

tion system, which would scram the reactor. It is not considered reasonable to assume that the worst possible sequence of pump failures would occur and that, in addition, the reactor protection system would fail to scram the reactor.

11-5.4 Extended loss of power. The decay heat removal system, described in Chapter 8, is designed to dissipate by natural convection enough residual reactor heat to protect the core from damage following a safety shutdown associated with the loss of all plant power.

The system is designed so that the plant, following such a shutdown, is completely safe for a period of 2 hr or less; if power is restored during this time, the plant is reoperable. Even after 2 hr there is no hazard to the personnel in the plant or the surrounding area; there is, however, the possibility of equipment damage (e.g., overheating electrical insulation). Battery and emergency diesel power of adequate capacity are available to prevent such equipment damage if the power loss continues after 2 hr.

11-6. Reactivity Accidents

11-6.1 Definition. Reactivity accidents are those that result in more excess reactivity being inserted into the core than is needed for normal rise in power. The result of such excess reactivity insertion is a too-rapid upward change in reactor power level, and if this rise were not stopped by the reactor protective devices, excessive local heating in the fuel elements could cause melting of fuel element cladding and the introduction of radioactive particles into the reactor coolant system. It will be shown in subsequent sections that even if it is postulated that all of the protection circuits fail, the reactor would be shut down, for any conceivable method of adding excess reactivity, by the bulk formation of steam in the coolant channels. This shutdown occurs through the cessation of neutron moderation and the consequent reduction of fissioning rate. In such a case extensive damage to the core could result, but all radioactive products would be contained in the reactor coolant system.

11-6.2 Plant design as related to reactivity accidents. The control systems of greatest interest in a study of reactivity accidents are the reactor rod control system and the reactor protection system.

Safety considerations used in the design of the rod control system are:

(1) Outward movement of control rods is always under the control of the reactor operator at the main control console.

(2) When it is necessary to provide for motion of control rods from an auxiliary panel (as for test purposes), permissive action is required of the reactor operator, and permission can be withdrawn at any time by the

reactor operator. Prominent means of monitoring all rod motion is provided at the main control console.

(3) Rod withdrawal rates are kept as low as possible consistent with the rate necessary to override the fastest xenon transient to be encountered.

(4) Malfunctions (such as overspeed of control motors) are limited in their ability to increase rod rates by inherent limitations in the equipment.

(5) Circuitry is such as to limit the number of rods moving at once, even with equipment malfunctions.

The reactor protection circuitry has been designed with the objective of preventing bulk boiling or burnout in the hottest channel and hence is designed to prevent any core damage. It is to perform this function in the face of predictable potential equipment failures and failure of the operator to exercise good judgment or to recognize the import of information given him. The system cannot, however, protect the core against bulk boiling or burnout for all hypothetically conceivable cases of simultaneous but unrelated malfunctions and maloperations.

A description of the rod control and reactor protection systems is given in Chapter 9. A listing of some of the features of those systems, which illustrate the manner in which these safety considerations have been incorporated into the plant design, follow:

(1) In-out control of all inverters except the spare bus inverters is located only on the reactor section of the console.

(2) Rods may be transferred to the spare bus inverters at the auxiliary panel only by having the operator at the control console place a selector switch in a permissive position. The spare bus inverters may then be positioned at the auxiliary panel if the reactor operator places the permissive switch in the proper position. By this switch he can also remove control at any time. Normal position for this switch in all procedure setups is "OFF." Motion of the spare bus inverters is transmitted to the full size dial position indicators on the console. An indicating light for each rod on the reactor section of the main instrument panel is illuminated when that rod is on the spare bus.

(3) Rod withdrawal rates are set so that at maximum worth approximately 25 sec are required to introduce sufficient reactivity to achieve prompt criticality.

(4) The commutator inverters will stop if either the field circuit or the armature circuit of their drive motors should open. If the field current should be reduced to a low value (not zero), the maximum speed obtained by the inverter would be 150% of normal full field speed. Each inverter motor is tested at the vendor's shop to verify this characteristic.

(5) The circuitry for transferring action from one inverter to another is by a snap action switch of the "either-or" type. If the switch contacts fail to move, the second inverter cannot be energized because the same contacts must be utilized in the new position.

TABLE 11-1

TABULATION OF SCRAM SIGNALS AND DELAY TIMES

Cause of actuation	Source of signal	Delay time (rods unlatched)
High coolant hot-leg temperature, 555°F	One of four loop resistance thermometers	125 msec
175% nuclear power level	Two of four compensated ion chambers	417 msec
Overpower at reduced flow	Two of four compensated ion chambers	425 msec
Flow reduction at fixed power	One or two of four pump power relays	342 msec
Loss of flow independent of power	Three of four pump power relays	375 msec
Low pressure, 1600 psig	One of two pressure instruments	125 msec

(6) All protection circuitry action that calls for an insertion of rods overrides all "rod-out" signals and blocks the "rod-out" relays.

(7) The scram relays simultaneously open the single scram circuit breaker, which removes all voltage to the rods, and the circuit breakers from the three generators supplying power to the rods. Either condition will cause rod release.

(8) For the most important protection signals—excess neutron flux level and loss of flow—two essentially independent scram systems exist, either one of which would produce the result even if the other failed. One consists of a chain of devices which operate directly on excess neutron flux level or directly on low flow to an independent output magnetic amplifier and relay. The other compares the flux with the flow and operates through another output circuit to produce protective action.

(9) Excessive neutron flux level is detected by four nuclear instrument chains. Any two of these reading above the scram limit can produce a scram.

(10) Excessive temperature is detected by four (one per loop) independent temperature instruments. Any one can produce a scram.

(11) Low pressure is detected by two independent pressure instruments, one on the pressurizer and one connected on a line to the reactor vessel which cannot be isolated. Either one can produce a scram.

(12) The above protection scram signals have the delay times, i.e., the time between the exceeding of the physical parameter limit and the actual rod movement, as given in Table 11–1.

(13) In addition to scram functions, circuitry is provided to insert rods at normal rod movement rates for two less severe abnormal values of plant parameters: excessive power-to-flow ratio (resulting from high power and/or low flow), and fast period during startup.

The potential severity of a reactivity accident has dictated the procedural techniques and automatic cutback or shutdown provisions of the control system. If proper procedures are followed, none of the cutback or shutdown signals are required to function for a reactivity type of accident. If errors of procedure are made, the automatic controls will function to prevent damaging accidents. However, in order to examine the effects of combinations of procedural and control system failures, various studies and analyses have been made. Most of these analyses employ the results of analog simulation of the reactor kinetics and the power plant heat-transfer characteristics from the reactor to the steam turbine. These analyses are discussed in detail in the following sections.

11–6.3 Specific reactivity accidents. *Excessive rod withdrawal.* In this accident it is assumed that the reactor control rods are withdrawn continuously. Rod withdrawal may have been initiated to make a desired outward adjustment of rods for correction of temperature when the reactor is operating at power, or to adjust the rate of power rise when the plant is being started up. It could also have been caused by a circuitry malfunction or by operator error.

It is not reasonable to expect core damage to arise from a continuous or excessive rod withdrawal accident occurring in the startup range, since the start-up rate cutback or either an excessive power-to-flow ratio or high power level scram would prevent it.

An excessive rod withdrawal accident occurring when the reactor is in the power range is more likely to be caused by an automatic control system failure than by an operator error. There is no requirement on the operator at any time to make outward adjustments of the rods of more than momentary nature. However, if such an accident should occur, it could conceivably occur with the rods at maximum worth. The accident was studied assuming the plant initially at full power and a rod rate corresponding to reactivity insertion of 2.5×10^{-4} Δk/sec. The results of the study show that a power-to-flow ratio cutback, a high power scram, or a high temperature scram would adequately protect the core from damage. Because three separate automatic protection features, any one of which would adequately protect the core from damage, would have to fail, and also because a manual cutback or scram on the part of the operator would

be adequate to protect the core, it is concluded that this accident is not a potential source of core damage.

Excessive heat withdrawal by steam demand. In this accident it is assumed that the turbine throttle valve or the controlled steam relief valves are opened by operator error or control circuit malfunction. This introduces reactivity into the core by lowering the temperature of the reactor coolant. Since the reactor normally responds to steam demand through this temperature effect, the situation becomes an accident only when excessive amounts of steam demand are imposed, or when a steam demand is made with the reactor critical but at a power level below the power range of operation. The latter case causes a rapid increase in power before the adjusting effects of the temperature coefficient are available.

Factors of importance in considering this accident are the procedures and limits on valve operation, the speed of valve opening, and the reactor protective circuitry, which would take effect if malfunction or maloperation introduced excessive reactivity.

The reactor protection system provisions for preventing damage from this accident, should it occur when the reactor is being started up and has not reached the power level for effective control by negative temperature coefficient, are (1) the fast startup rate cutback, (2) the power-to-flow ratio cutback, (3) the power-to-flow ratio scram, and (4) the high nuclear level scram.

If the reactor is operating in the normal power range and the controlled steam relief valves (which, combined, have a possibility of imposing an 80% power load on the reactor plant) open suddenly and improperly, the power-to-flow ratio cutback would cut back power with little transient effect in the reactor. As backup, scrams on high power-to-flow ratio or high nuclear level would prevent any damage to the reactor. Steam valve malfunctions with the reactor at power are of minor consequence to the reactor. Limitations of the heat transfer capability of the boilers prevent overpower to any serious degree.

Plant operating procedures require that all four main steam stop valves be closed during all normal startups. These valves are motor operated and require 60 sec to open. They are not opened until the reactor is at the beginning of the power range of operation, about one percent power. Should one or more start opening during the time the reactor is in the subpower range, the turbine throttle would still be closed. If other valves were open, the long time required to open the main stop valve gives ample warning to both the reactor and steam plant operators to forestall a reactor transient.

For damage (hot spot burnout or melting) to occur from excessive heat withdrawal by steam plant valves the following events would have to transpire:

(1) The main steam stop valves would have to be open during a reactor startup—a violation of proper procedures which would have to be overlooked by at least two operators during prestartup checkout.

(2) The circuitry for controlled steam relief valves (four separate valves totaling 80% capacity) would have to fail, causing them all to open at the specific time of criticality.

(3) The fast startup rate cutback would have to fail to function.

(4) The low-pressure scram would have to fail.

(5) The power range power-to-flow ratio cutback would have to fail.

(6) The power range power-to-flow ratio scram would have to fail.

(7) The power range high power scram would have to fail.

If all these events transpired, a nuclear runaway would not result; rather, the power level would overshoot, possibly causing some melting of fuel elements, but all of their products would be contained within the coolant. Thus, it is concluded that a damaging accident from this cause is not feasible.

Cold water introduction by startup of a reactor coolant loop. In this accident it is assumed that, through operator error combined with circuit malfunction, a nonoperating loop containing water colder than that in the operating loops is put into service by opening its valves and turning on its pumps. An accident may be created because the colder water introduces excess reactivity, through the medium of the negative temperature coefficient of reactivity, and a fast rise in power may result.

Of importance in considering this accident is the temperature difference existing between the operating and nonoperating loops, the plant procedures for placing a loop in service, and the reactor protection circuitry which would take effect if malfunction occurred.

Two independent protective means are provided for preventing "cold water accidents" in the event that the operator violates approved procedures. The first device, the temperature difference interlock, measures the temperature of the water in the boiler of the out-of-service loop and compares it with the highest reactor coolant temperature in any of the lines from the boilers to the reactor. If the difference is greater than 20°F, a relay fails to energize and blocks the starting of the pump in the loop being put into service. The device is designed to fail in the blocked position on loss of power to the measuring circuits. To permit warmup of a cold loop when both main coolant valves are in the fully closed position, the interlock on the pump can be overridden. The second device, the main loop valve opening interlock, is an interlock on the safety shutdown circuit breaker (rod control system). It prevents opening of the main coolant valves of an inactive loop when that circuit breaker is closed. Thus once the valves are closed, a loop cannot be put into service while the reactor is critical.

The probability of the blocking function of the temperature interlock being exercised during pump startup is extremely unlikely. The operator must first put a loop selector switch into a position corresponding to the loop being placed in service. Until this is done, the pump starting controls are inoperative. This switch is located immediately below the four loop cold leg temperature recorders and the four loop hot leg temperature recorders. Thus, by the nature of the operational procedures, a pump startup cannot be casual, and the operator will be reminded to check operating temperatures of the loops.

The "cold water accident," for purposes of analysis, may be considered one in which flow in one loop, initially operating at full power with the other loops, is suddenly reduced to zero by stopping its pump and then is restored by restarting the pump after about two minutes have elapsed, at which time the coolant water in the boiler will be at its minimum temperature. The temperature interlock is assumed to have failed to prevent starting the pump. In addition, the operator must have disregarded proper operating procedures, preventing restoration of a loop to service with the reactor critical, and must have ignored available information, presented directly in front of him, indicating that a loop has not been properly warmed up.

When the reactor coolant flow to a boiler is stopped, the boiler can no longer deliver steam. The steam bubbles, which occupy an appreciable percentage of the volume on the secondary side of the boilers, collapse and the secondary water level falls. The feedwater system restores the level with approximately 325°F water. Analysis (based on boiler operation at full power at the time coolant flow is stopped) indicates that the heat withdrawal by the boiler secondary water will drop the temperature of the reactor coolant a maximum of about 52°F below that of the other loops. Only the coolant within the boiler will be so affected. It takes approximately one to two minutes for this minimum temperature to be reached. If the main coolant stop valves have not been closed, the coolant in the boiler will be warmed up—after the minimum temperature has been reached—by backflow through the hole in the main loop check valve. If the pump is started again while the boiler coolant temperature is at its minimum, this cooled reactor coolant will move out of the boiler and to the reactor. Approximately 3 sec are required to move all of the low temperature reactor coolant out of the boiler and to cause the temperature of the water leaving the boiler to rise again to the normal cold-leg temperature of 508°F.

Analysis indicates that for short-time transients, one quarter section of the core is virtually independent of the remaining three quarters, for the following reasons:

(1) Experimental investigation has shown that very little mixing occurs

in the reactor inlet plenum, so that most of the cold water entering the vessel from one loop travels straight up through the core in a localized (one quarter section) area.

(2) The neutron flux coupling characteristics of the large seed and blanket type core are loose, so that reactivity disturbances in one area of the core can initially produce nuclear excursions independent of the other areas.

In analyzing this accident, calculations were made to determine the power and core metal temperature excursions. This was done for various temperature differentials (ranging from zero through 200°F) between loops at the vessel inlet. Results are very conservative because the temperature reduction of all the water in the seed cluster was assumed to be instantaneous. This conservative analysis estimates that a temperature differential of about 52°F produces a reactor period of approximately eight seconds, which is larger than that required for prompt criticality. It is not known from available data whether or not the phase transformation temperature (1111°F) of the seed fuel would be exceeded. However, the melting temperature (3320°F) of the seed fuel would definitely not be exceeded. Experiments at the Naval Reactor Facility at Idaho Falls, Idaho with reactor period as low as 0.080 sec, corroborate the conservatism of the PWR calculation on this point.

Phase transformation is a change in the basic crystalline structure of the seed fuel which results in a volume increase of the fuel material when the phase transformation temperature is exceeded. The phase transformation temperature is used to determine the temperature difference interlock setting; this is accomplished by defining the maximum temperature differential the reactor can withstand without the seed fuel passing through phase transformation. The amount of volume change is dependent upon the magnitude and duration of the excessive temperature. For the rapid type of temperature transient (52°F) encountered in this accident, the seed fuel volume change is relatively small. Therefore, little, if any, rupturing of the fuel element cladding is anticipated. Thus, it is apparent that no potentially hazardous condition (significant release of fission products to the coolant) can arise from stopping a pump in a reactor coolant loop and starting it again when the coolant water in the boiler is at its minimum temperature during the ensuing transient.

Summarizing, for this "cold-water accident" to occur, (1) a main coolant pump must have been stopped during full power operation, then (2) the operator, improperly, must have restored it to service at the time when the boiler coolant temperature was at its minimum value during the ensuing transient, and (3) the temperature interlock must have failed to prevent the action in spite of a temperature differential greater than 20°F. With this sequence of events and failures, the seed fuel temperatures

might exceed the phase transformation temperature but would not exceed melting temperatures. Thus, it is concluded that no significant release of fission products to the coolant can arise and thus that no hazard exists from this accident.

Xenon burnout without reactor control. In a xenon burnout accident it is assumed that the reactor power level is raised to or near rated value following a shutdown in which Xe^{135} has accumulated to a high or maximum value, and that the automatic control system and the operator fail to make compensating adjustments as the xenon concentration is reduced by neutron absorption. Of importance in considering this accident is the maximum possible concentration of Xe^{135}, the rate of reduction by neutron flux, the effectiveness of the control rods, and the circuitry that would protect the reactor should such an accident occur.

The reactor rod control system has been designed to provide a rate of rod motion sufficient to enable the control system to compensate for the rate of reactivity insertion by xenon depletion at maximum neutron flux level. The theoretical maximum xenon burnout rate is obtained when the power is increased in a step from 0 to 100% after a shutdown of approximately 10 hr, the time required for maximum xenon to accumulate. The shutdown would have to be preceded by a period of at least 50 hr at full power.

If the automatic control system became inoperative at this particular time, and the operator were not aware of this, a high temperature alarm would sound. If the operator failed to act upon hearing the alarm, a high power-to-flow ratio cutback or a high temperature scram would protect the core. If the operator failed to act and neither the cutback nor the scram functioned, the temperature would continue to rise and boiling would occur in the hot channels. In order for this accident to continue even to the point of requiring automatic corrective action, one must postulate a complete failure of both the reactor and steam plant operators and of the chief operator to respond to the Station alarms that would be actuated.

Plant operating limitations make this rather mild transient a very remote possibility. Operating procedures require about 30 min for fully loading the turbine. Any step increase in power to 100% from 0% would be corrected at once by closing of the throttle. Loading the reactor in steps to 100% power over a one-half hour period would reduce the rate of xenon reduction by burnout to a considerable degree. Even if the incident should occur, the many plant instruments which would show the relatively slow rise in temperature to both the reactor plant and steam plant operators would give ample opportunity for corrective action. Such corrective action would consist of (1) inserting rods manually to compensate for xenon reduction, (2) reducing plant load at the turbine throttle, or (3) inserting rods by scram. The last method should never be required for this condition since either (1) or (2) would be adequate.

It is therefore concluded that it is unreasonable to assume that this accident would be allowed to continue to the point of producing any damage to the core.

11–6.4 Infeasible reactivity accidents. The preceding discussion has been concerned with accidents that are caused by varying degrees of maloperation and equipment malfunction, with three or more unrelated failures required to produce damaging conditions. It is concluded that while any one of the events discussed above may be feasible, combinations of them are not. Moreover, none of the accidents listed above results in more than contamination of the reactor coolant system.

For the sake of completeness, a discussion is included here of two effects of combinations of operator malfeasance and equipment malfunction which might be thought to lead to severe core damage.

Simultaneous withdrawal of control rods. As previously pointed out, provision is made for the operator to allow limited control of individual rods, or a group of four rods, from a control point other than the normal programmed control switch. This is done to test the rod operation by allowing rods to be moved by a spare inverter. The operator can assume control of this function at any time and has full instrumentation to apprise him of the actions being taken when control is granted to the auxiliary control board operator. This function is normally carried out on one rod at a time.

The spare inverters are fixed by built-in gear ratios to operate at one half the speed of the normal inverters. Thus, the maximum rate of reactivity insertion obtainable (if four rods are on the inverter) is 1.5×10^{-4} Δk/sec. It is possible, although no procedure or necessity calls for it, to put more than four rods on one spare inverter. (This is not the case for the regular inverters.) As soon as a fifth transfer is attempted, the circuit is overloaded to the point where it is questionable whether the rods will remain latched. It is certain that they will unlatch when the seventh transfer is attempted. This, of course, shuts down the reactor. Since eight rods could be moved at spare inverter speed without excessive reactivity rate, it is concluded that a dangerous excessive reactivity insertion rate cannot be obtained from spare inverter operation.

The operator may also remove control of a regular inverter for a group of four rods from the program sequence and transfer this control to the main console. Thus, this control is available only to himself. Two such inverters may be so transferred, allowing two groups to be controlled in this manner. The purpose of this function is to allow the operator to trim the positions of the rods a group at a time to align them evenly, or to shift the programming in accordance with changes in xenon conditions in the core. Such occasions would never arise during startup, and no operating

requirements would call for simultaneous operation of more than one group. Since the rods are required to be in programmed sequence for normal operation, this function would be exercised briefly, and the rods returned to program. There is no combination of circumstances which could conceivably lead the operator, even in panic, to operate both of the special control switches, each for a rod group, simultaneously with the normal control. Even should he desire to do so, he could not operate all the switches, since all rod motion switches are spring returned to "OFF."

A simultaneous motion of three groups of rods would result in a maximum reactivity insertion rate of 9×10^{-4} Δk/sec if all groups were at the point of maximum effectiveness. A rate of 10×10^{-4} Δk/sec, if applied continuously from below criticality to one percent power, would result in a reactor period of 0.066 sec. A period of 0.005 sec or shorter would be required to melt the fuel plates. Thus, even this implausible accident would not cause a core vaporization or a release of fission products beyond the site boundary.

Startup of a reactor coolant loop at reduced temperatures. If a loop is inoperative for maintenance, it may be at any reduced temperature down to the ambient temperature of the plant container. The reactivity transients that would result from injecting cold water from an inactive loop into a hot operating reactor indicate that for temperature differentials at the reactor vessel inlets of less than 160°F, periods longer than five milliseconds would result, and the core fuel would not melt or vaporize. For temperature differentials at the vessel inlets greater than 160°F, the probability and extent of melting or vaporization of the core fuel increases with increased temperature difference. Consequently, in view of the potentially severe hazard presented by this type of accident, extreme operational procedures have been specified and devices have been installed to prevent such an accident.

Before an isolated loop is restored to service, operating procedures and interlocks require that the loop water temperature be brought up to approximately that of the remainder of the plant. The cold loop is warmed up with its main stop valves closed. One way of accomplishing this is to open the bypass valve of the cold loop and establish flow with the loop pump at one-half speed while a small interchange flow between this loop and an operating loop is maintained through two-inch interconnecting lines. The control circuitry is so designed that the main stop valves cannot be opened to bring the loop into service while the safety shutdown circuit breaker of the rod control system is closed. Thus, at the time a loop is brought into service, the breaker must be open and the control rods must be fully inserted so that the reactor is shut down. If, as a result of some operator or equipment failure, the main loop valves were opened, the pump could not be started at either speed with the cold loop more

than 20°F colder than the operating loops, because of the temperature interlock.

To arrive at the point of attempting to turn the pump on (at which it is assumed that the temperature interlock has failed), the operator would have to proceed contrary to proper operating procedure and in disregard of the information on the meters immediately above the loop selector switch, which show the hot leg and cold leg temperatures of all loops. He would then have to position the loop selector switch to the loop that is at reduced temperature and then operate the pump switch for that loop.

Inasmuch as two equipment failures and four instances of operator negligence must occur before the reduced temperature water could reach the core, the cold water accident in which a loop at greatly reduced temperature is brought into service with the reactor critical is not considered a credible casualty.

11–6.5 Conclusion that no feasible reactivity accident can result in damage to fuel elements. The maximum fuel center temperature attained was calculated for two sets of postulated rod withdrawal accident conditions. In one of these, typical hot reactor conditions at 525°F and 2000 psia were assumed; in the other, typical start-up conditions of 190°F and 100 psia. The calculations demonstrate that, for both cases, an initial reactor period of 0.003 sec or shorter is required for vaporization of seed plates and an initial period longer than 0.005 sec will not result in a fuel plate temperature that reaches the melting point of the metal.

It has been calculated that no feasible reactor accident involving rod withdrawal will result in an initial period smaller than 0.3 sec. Since initial periods of 0.005 and 0.003 sec are required for melting and vaporization, respectively, no reactivity accident involving control rod motion will melt or vaporize fuel elements.

In the infeasible case of simultaneous withdrawal of three groups of rods (requiring three hands on the switches), it was pointed out previously that the minimum initial period obtainable would be 0.066 sec. Thus even this accident would not damage the fuel elements and release fission products.

Cold water entering the core would have to be about 160°F, or more, colder than the water in the core to result in a reactor period of 0.005 sec. During normal operation with four loops in service, as discussed above, no plant transient can result in temperatures more than about 52°F below those in the core, and the plant has been designed to prevent the introduction of this colder water from the boiler tubes into the reactor. Therefore, it is concluded that no feasible cold water accident can melt or vaporize the fuel elements and release fission products.

11-7. Core Handling Considerations

The potential hazards associated with core handling were thoroughly investigated. Core handling equipment and techniques described in Chapter 4 were designed not only to prevent mechanical damage and protect the large investment, but also to make negligible the possibility of a nuclear incident even in unexpected circumstances.

11-7.1 Storage and shipping. Special racks are provided for shipping and storing fuel assemblies. The spacing of assemblies in these racks will not permit a critical mass. No geometry of dry (unmoderated) seed and blanket assemblies can provide a critical mass. Seed assemblies are shipped separately from blanket assemblies. Blanket assemblies present no criticality problem. No seed assembly is handled or moved unless a control rod or its equivalent has been inserted into it. No criticality hazard is possible if the integrity of the shipping racks is maintained. A criticality hazard could result only under the highly improbable condition that four seed assemblies without rods or seven with rods break out of their positions in the shipping rack and fall, as a compact contiguous array with axes aligned, into a body of water sufficiently large to immerse them.

The shipping route of the Core I fuel assemblies was examined carefully in advance and the actual shipment was accomplished under carefully controlled conditions so that no feasible accident en route to Shippingport could have resulted in forming a critical assembly of fuel material in water.

11-7.2 Core assembly and insertion. Core I assembly was done in a dry area under maximum security conditions and thus presented no nuclear accident problem. A completely assembled dry core, even with the rods withdrawn, would not be critical, nor would a complete seed array immersed in water with the rods withdrawn. A completely assembled core immersed in water can be kept subcritical by the control rods; no nuclear poison is required in the water.

The assembled Core I was moved down the canal from the assembly area to its place in the reactor vessel. This operation was considered to present no nuclear hazard, because it was done with the canal empty and with all the control rods locked in place by a special safety hold-down plate.

11-7.3 Refueling. Refueling will be accomplished with the canal filled with water. The refueling equipment has been designed to prevent any conceivable nuclear accident. Nothing can occur during installation or removal of a single fuel element that can cause a nuclear incident. No operation involved in removing a core increases its reactivity. The core cannot become critical when the control rods are locked in by the safety

hold-down plate. Only a violation of procedures can result in a hazard. Refueling operations will proceed under close supervision to assure adherence to all core handling rules and procedures.

11–8. Waste Disposal System Considerations

The radioactive waste disposal system (described in Chapter 10) is designed to discharge effluents continuously in a manner such that the activities of the effluents are increased above the original activities of the air and river water by no more than one tenth of the maximum permissible concentration (MPC) values given in National Bureau of Standards Handbook 52. The design is such even though it would appear to be acceptable (Reference: NBS Handbook 61) to operate on the premise that only the concentration *average* over a year should be at or below one tenth of the MPC values. The operation of this system was approved by the Commonwealth of Pennsylvania.

Two extremely improbable accidents were analyzed to illustrate the maximum hazard that might result from operational error or equipment malfunction in this system. In the first accident, it was assumed that a rupture of one of the gas storage tanks in the waste disposal building occurs with an instantaneous release of its contents; in the second, it was assumed that an operator empties one surge and decay tank to the river in violation of established operating procedures.

The first accident is believed improbable because all process vessels in the system were built to ASME code standards and are equipped with safety devices to ensure against overpressure stresses. However, even if the accident did occur, no significant radiation hazard would be present at any reasonable distance from the point of release. Since the dose would result from noble gases, it would constitute an external hazard only.

The emptying of a surge and decay tank into the river is improbable because it would require a violation of established operating procedures. Monitors are provided on the effluent lines to ensure that the activity levels of the liquids and gases leaving the plant do not exceed the disposal criteria given above. Such an accident would require a period of about 16 hr. For a set of extremely pessimistically assumed conditions, the activity of the plant effluent at the point of discharge to the Ohio River would be only about 1.8 times the MPC value given in NBS Handbook 52. Subsequent dilution with the river water would reduce the effluent to well below tolerance levels before it reached its first point of use, about one half mile downstream.

Therefore, it is concluded that the radioactive waste disposal system offers no hazard to the area surrounding the Shippingport plant.

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CHAPTER 12

ELECTRICAL AND MECHANICAL COMPONENTS

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ELECTRICAL AND MECHANICAL COMPONENTS

12–1. Electrical Components*

12–1.1 General requirements. Whenever practical, conventional and commercially available electrical components were used in the Shippingport reactor electrical systems. In many cases, however, reactor plant requirements of extreme cleanliness and hermetic sealing dictated the use of special components. Many of the components were to be located in chambers where they would be inaccessible for long periods, so rugged equipment with a long, reliable, maintenance-free life had to be procured, or in many cases, developed. Stainless steel was used for its corrosion resistance, and class H electrical insulation (200°C hot-spot rating) was used for components adjacent to hot parts of the plant. Vacuum tubes are used only in nonvital circuits not affecting the safe operation of the plant, and even then in readily accessible locations only.

Because of the wide variety of complex electrical equipment, every effort was made to debug and test as much of it as possible at the vendors' plants. In some cases equipment made by one vendor was delivered to another vendor producing related equipment for composite testing before delivery to the site. Applicable equipment was tested for one week for aging and elimination of initial drift. These procedures greatly expedited final assembly and checkout of the plant by reducing the work to be done after the equipment was installed.

The electrical equipment can be divided by type into two general groups, nuclear and non-nuclear. These groups can each be subdivided, by function, into indicating and control equipment.

12–1.2 Nuclear instrumentation. The nuclear instrumentation system consists of four identical instrumentation channels, designated channels A, B, C, and D. These channels simultaneously and independently monitor the reactor neutron flux from source level to 175% of full power output. Each channel has three ranges: source, intermediate, and power (see Fig. 12–1).

The equipment consists of cabinets of electronic circuitry and six detector assemblies. The cabinets contain the components for channels A, B, C, and D, and the test equipment.

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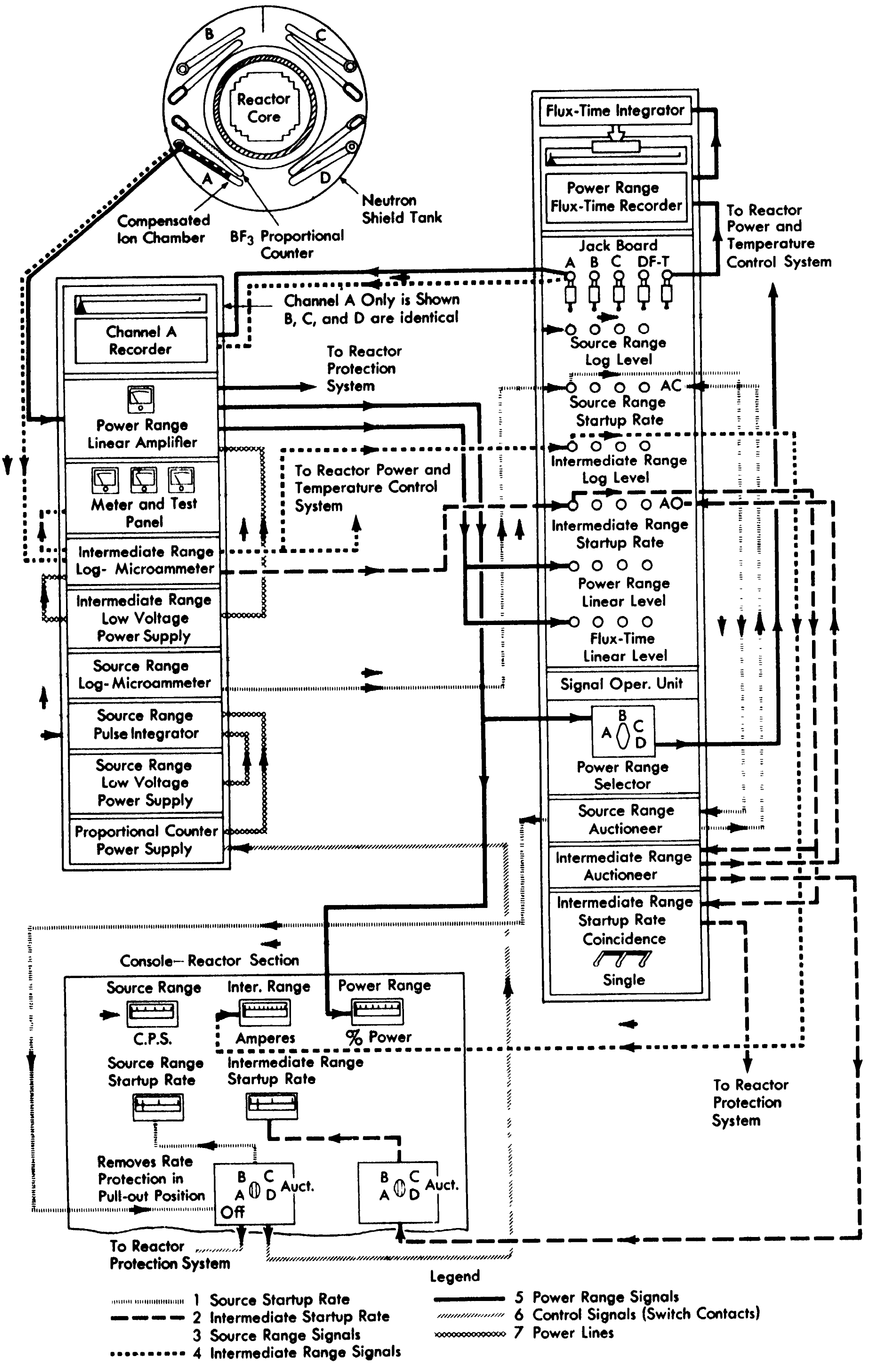

FIG. 12–1. Nuclear instrumentation system block diagram.

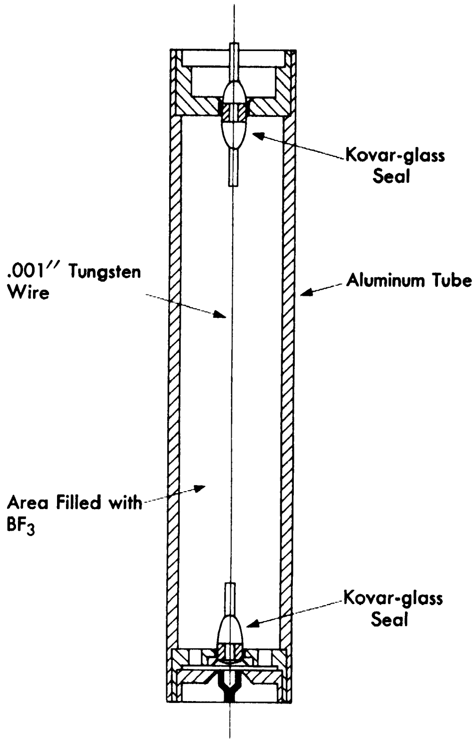

FIG. 12–2. BF_3 proportional counter.

Neutron detector assemblies in instrument wells in the neutron shield tank are connected by coaxial cables to the electronic circuitry contained in the cabinets in the control room. Two detectors are used per channel. A boron trifluoride proportional counter, commonly called a BF_3 counter, covers the source range, while a single compensated ion chamber provides the input to both the intermediate and power ranges.

A BF_3 counter is an instrument for detecting low level thermal neutrons. The counter, shown in Fig. 12–2, consists of a pure aluminum tube filled with boron trifluoride gas and has a very fine tungsten wire stretched along its center. The ends of the tube are sealed and the tungsten wire is electrically insulated from it. A high DC voltage is applied between the tube and the central wire. When a thermal neutron enters the counter it ionizes some of the BF_3 gas; these ions are attracted by the charge on the center wire, creating small pulses of current.

These pulses go to a pulse integrator which first amplifies the pulses and rejects the small pulses due to noise and gamma flux; it then integrates the pulses to produce a direct current output which is proportional to neutron flux level. This direct current goes to the *source range log microam-*

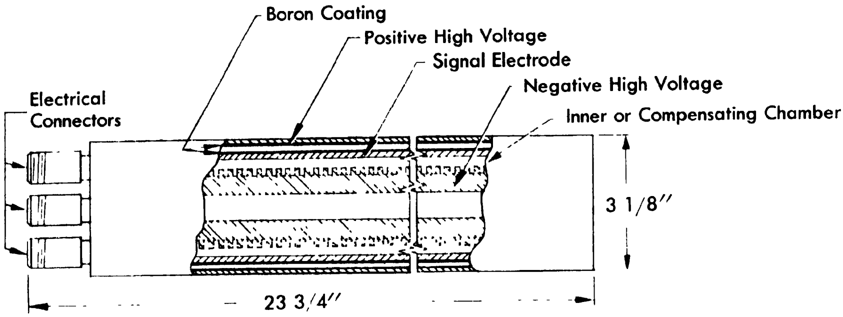

FIG. 12–3. Compensated ionization chamber.

meter circuit which indicates neutron flux level, in terms of counts (pulses) per second, on a logarithmic scale with a range of 1 to 100,000 counts/sec. The log microammeter also provides a second signal which is proportional to the rate of change of the log level, indicating the startup rate (from −0.2 to +2 decades/min).

Compensated ionization chambers are used to measure neutron flux during operation in the intermediate and power ranges. In principle they are similar to the BF_3 counters, but instead of being filled with a boron gas, the inside wall of the tube is coated with a boron isotope. The high neutron fluxes encountered in the intermediate and power ranges, impinging on the boron coating, create current pulses in such rapid succession that they run together and result in a continuous flow of direct current.

The boron isotope coated detector is sensitive to gamma radiation as well as neutrons. To compensate for the error caused by gamma radiation, a similar chamber is built inside the coated chamber (see Fig. 12–3). This concentric chamber is not coated and therefore is sensitive to gamma radiation but not to neutrons. The output signal from the second chamber is subtracted from that of the first, and the resultant signal is proportional to neutron level.

An *intermediate range log microammeter circuit* receives the DC signal (10^{-11} to 10^{-4} amp) from the compensated ionization chamber and provides both (1) a DC output for indicating the neutron flux level on a logarithmic scale and (2) a signal which is proportional to the rate of change of the level for indicating startup rate (from −0.2 to +2 decades/min). A third output, a signal produced when the startup rate exceeds a predetermined value, is provided for use by the reactor protection system.

The *power range instrument* takes a signal from the compensated ion chamber and supplies DC output signals for indicating reactor power level from 1 to 175% full power on a linear scale, for driving a flux-time integrator, and for use by the reactor protection system and the power and temperature control system.

12-1.3 Core temperature and flow instrumentation. A large number of temperatures and flow rates are measured in various parts of the core principally to gain information that will be useful in future designs. Water and metal temperatures are measured by thermocouples; flow is measured by differential pressure cells across various channels in the core. Those thermocouples in the reactor core are chromel-alumel to withstand the high fuel plate temperatures. The leads for these thermocouples run from the fuel plates or flow channels to the head of the reactor vessel through the reactor coolant. Therefore, they are insulated with compacted or swaged zirconium oxide inside stainless steel sheathing. The hot junction is joined to the end of the sheath to provide fast response to temperature changes. The sheaths are welded into a tube sheet at the head of the reactor.

The method used to measure coolant flow through the core is described in Chapter 4. The flow signals are converted to differential pressure signals by means of differential pressure (D/P) cells. The cells are mounted

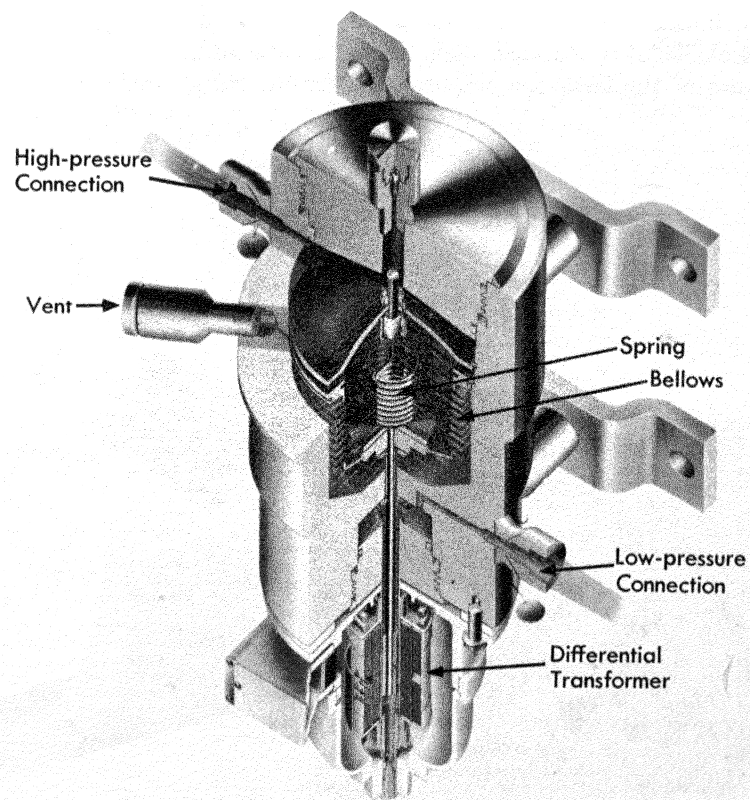

FIG. 12-4. Differential pressure cell.

on a trellis on the head of the reactor vessel (see Fig. 9–4). The cells use a spring and a metallic bellows to position a movable magnetic core in a differential transformer (see Fig. 12–4). The bellows moves until the differential pressure force is balanced by the force of the spring. Thus the transformer core is positioned so as to cause an electrical output proportional to differential pressure and hence to coolant flow.

12–1.4 Plant instrumentation. The *temperature* signals used in controlling the reactor portion of the plant are obtained from special resistance thermometers designed to produce a large signal and have a fast response. Each thermometer consists of a molybdenum wire resistance element surrounded by aluminum oxide and swaged inside a stainless steel sheath. The leads are brought out of this sheath through ceramic-to-metal seals. The element is bent into a U-shape and mounted so that it projects through a hole in the side of the pipe where the temperature is to be measured. A protective cage with holes in it surrounds the U-shaped element to reduce vibration caused by the water flowing past it and protect it from impact. Before calibration and installation, these elements are soaked at 700°F to stabilize them and eliminate initial drift. Changes in resistance of the leads are compensated for by using three leads, two of

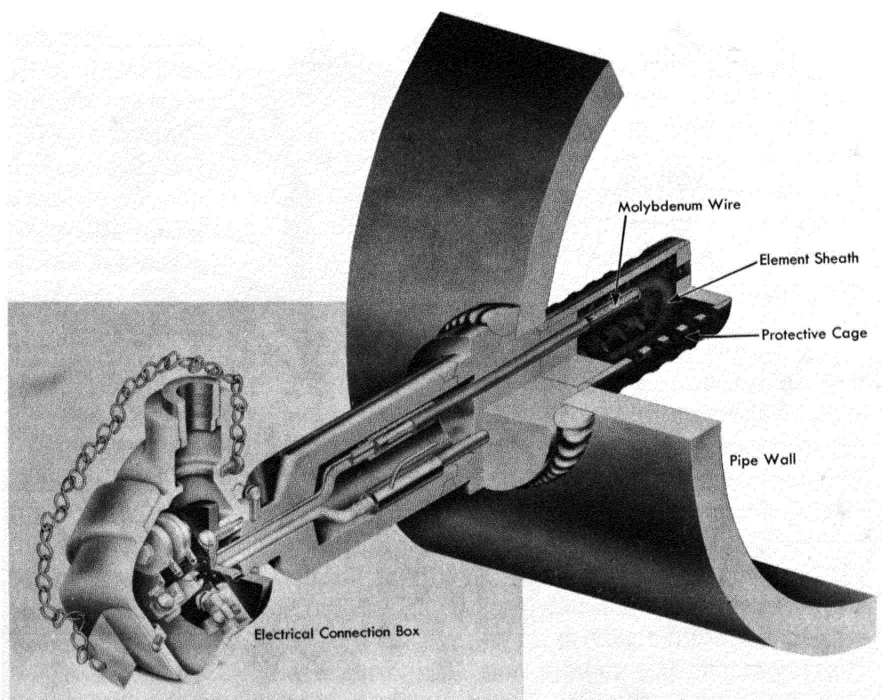

FIG. 12–5. High speed resistance thermometer.

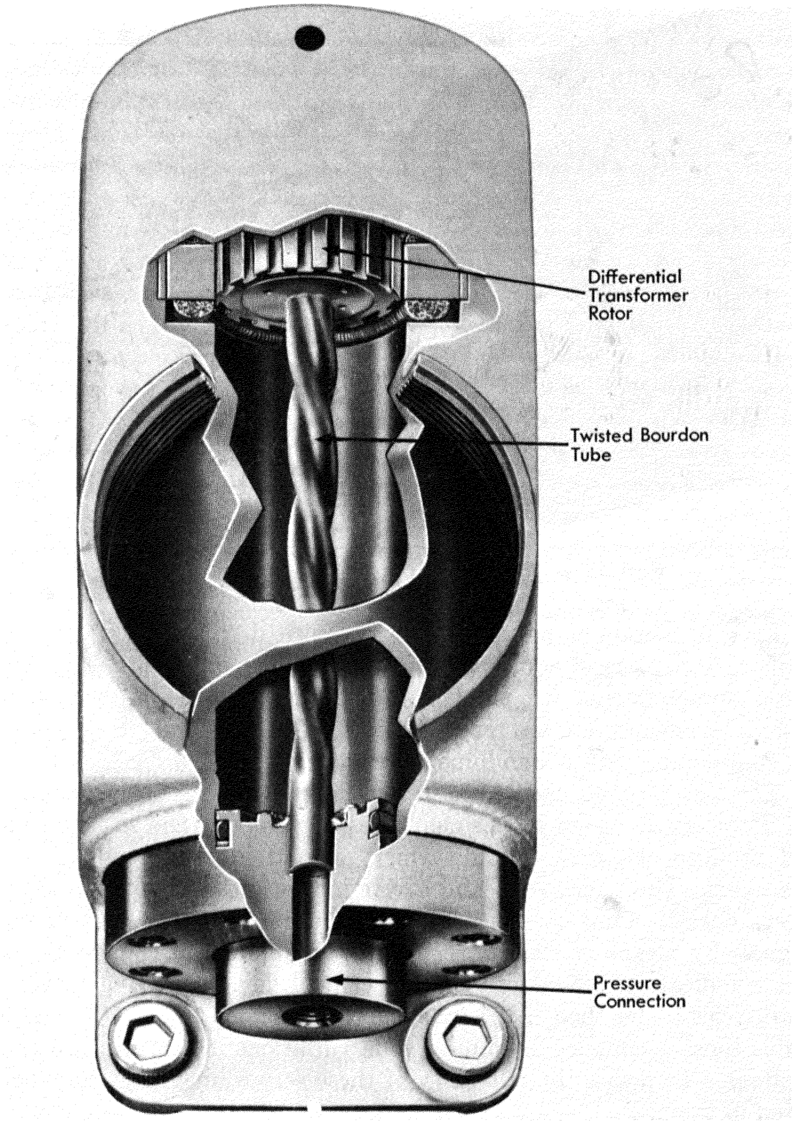

FIG. 12–6. Pressure transmitter with twisted Bourdon tube and rotary differential transformer.

which are connected together at the temperature sensing element. This thermometer assembly is shown in Fig. 12–5.

Water pressure measurements to be used for control purposes or pressure measurements made on water suspected of containing some radioactivity are accomplished by a twisted bourdon tube coupled to a rotary differential transformer as shown in Fig. 12-6.

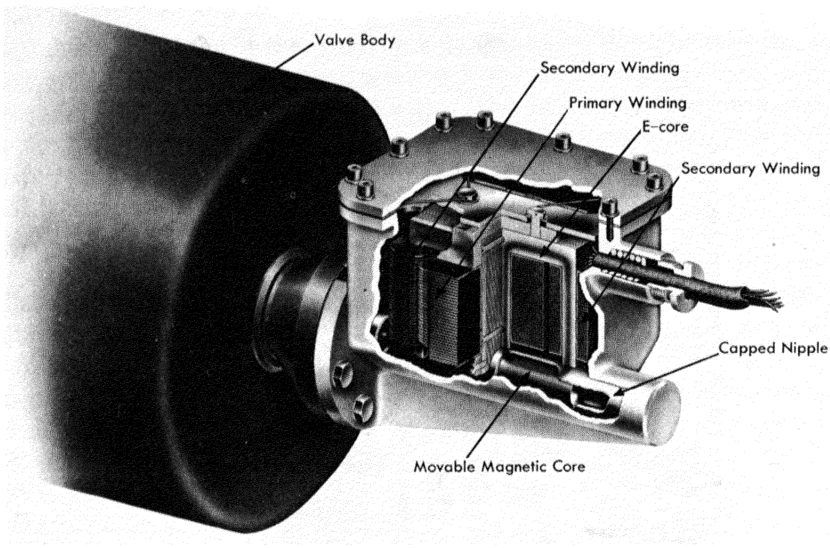

FIG. 12-7. Valve position indicator sensing element.

Flow is measured by measuring pressure drop across a venturi, an orifice, or a piece of equipment such as a boiler or pump. The pressure drop measurements for plant instrumentation purposes, like those for core instrumentation, are made by D/P cells. Similar D/P cells are used in conjunction with a standpipe for measuring water levels.

The *amplification and indication equipment* used for the different types of detectors discussed in this chapter is all similar. The sensing elements all produce electrical outputs which are fed into null balancing servo receivers. These receivers can provide indication of the measured parameter, operate limit switches at predetermined points, and retransmit signals by means of slidewire, synchro, or differential transformer to remote indicators, recorders, or other control devices. Using servos gives accuracies better than $\pm 1\%$ (in some cases as good as $\pm 0.1\%$) and maintains these accuracies in spite of wide variations in temperature and line voltage. To insure the reliability of these servos, magnetic amplifiers are used in them.

A number of *hydraulic cylinder operated valves* are used inside the various inaccessible chambers. Whether these valves are open or closed must be known at all times, even when there is no flow (for example, to avoid damaging a pump by starting it when its inlet and/or discharge valve is closed). A remote valve position indicator is used for this function. It consists of an *E*-core differential transformer mounted around a capped nipple extending from the body of the valve. Inside the nipple is a movable magnetic core, connected to the operating piston and moving with it.

The output of the differential transformer is determined by the position of the movable core. The low level output from this device is fed to a magnetic amplifier which operates a relay indicating position by means of pilot lights. This set-up is also used to provide permissive signals for control purposes. The differential transformer assembly is shown in Fig. 12-7.

The amount of hydrogen dissolved in the reactor coolant is measured by a *palladium diffusion gage*. Palladium has the unique property of allowing hydrogen, and nothing else, to diffuse through it, particularly at temperatures above about 600°F. A small flow sampling line is connected in parallel with the reactor coolant system and passes the pressurized water through a heater and a chamber which has a partition of palladium in it. The amount of hydrogen that diffuses through this partition is proportional to the dissolved concentration, the temperature, and the pressure. The pressure on the hydrogen side of the palladium barrier is measured and indicates the dissolved hydrogen concentration.

12-1.5 Rod control. Reluctance motors on the rod drive mechanisms are used for rod control. Power for these motors is obtained from inverters which produce a three-phase, AC output adjustable over a frequency range from 0.85 cps down to and including 0 cps. The inverters are actually motor-driven, high power, three-phase sine potentiometers fed from a DC bus consisting of three 45-kw motor-generator sets capable of carrying 100% overload for 30 min. The inverters have an efficiency of about 80%.

There are ten inverters, two of which are spares; the other eight each supply four rod drive motors. Operating the rods in groups of four provides sufficient flexibility of control in normal operation. The spare inverters can be used to position fewer rods if special effects are desired. For connecting each rod drive motor to its normal or spare inverter, there is a transfer device which consists of two contactors and a transfer switch combined to provide make-before-break operation for both downdrive and updrive. This prevents open circuit switching from scramming the rods, since the rod drive mechanisms are designed with very short drop-out times to reduce scram time delay. Switching from one inverter to another is always done at 0 cps to further reduce the possibility of dropping a rod. Before switching, the two inverters must be synchronized; synchronization is indicated by a synchro differential connected between the two inverters.

Two separate devices are used to indicate the positions of the control rods. The first, a reluctance type indicator operating in parallel with the rod drive motor, gives highly accurate position indication provided it is set at zero when the rod is at the bottom of its travel and the rod and indicator move in unison thereafter. The other, a transformer device, is less accurate but is not ambiguous.

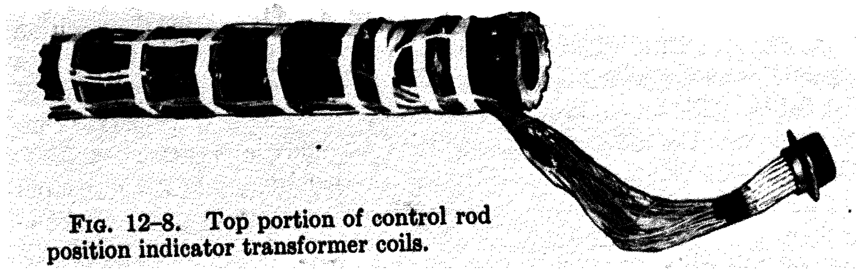

FIG. 12-8. Top portion of control rod position indicator transformer coils.

The reluctance type consists of a synchro receiver connected to the rod power supply in parallel with the rod drive motor so that it rotates in the same way. The synchro drives a pair of pointers through a gear train whose ratio is selected to turn one pointer one revolution per in. of rod travel and the other pointer slightly less than one revolution for the complete rod travel. The pointers and gear train constitute a very small load on the synchro and cause negligible error, so that the accuracy of this indicator is essentially the same as the accuracy with which the mechanism positions the rod.

The other indicator is an arrangement of small transformers (shown in Fig. 12-8) mounted on the rod drive mechanism above the rod drive motor. Each one is three in. long, and 24 of them cover the 72-in. rod travel. The mechanism lead screw acts as a core for these transformers, so that when the rod is at the bottom, there is no core in any of them. When the rod is being raised, the mechanism lead screw moves up through the centers of the transformers, increasing their couplings and secondary voltages one by one. Each secondary is connected to an incandescent indicator lamp. When the magnetic lead screw is within a transformer, the corresponding lamp is lighted, but when the lead screw is below the transformer the secondary voltage is so low that the lamp will not light. This large voltage change is the result of changes in both the mutual inductance and the self-inductance of the primary, which is operated on a constant current supply. The 24 indicator lamps for each rod are mounted on the main instrument panel (shown in Fig. 9-18) in a vertical row behind a red filter. In operation they provide a graphical representation somewhat resembling that of a red alcohol thermometer. Because this indicator detects the position of the lead screw, which is permanently attached to the control rod, its readings are not subject to ambiguity as is the case with the reluctance indicator. However, because there is only one transformer and light for each three inches of travel, its accuracy is limited to $\pm 1\frac{1}{2}$ in.

12-1.6 Reactor plant control equipment. To obtain maximum reliability in the power and temperature control system and the reactor pro-

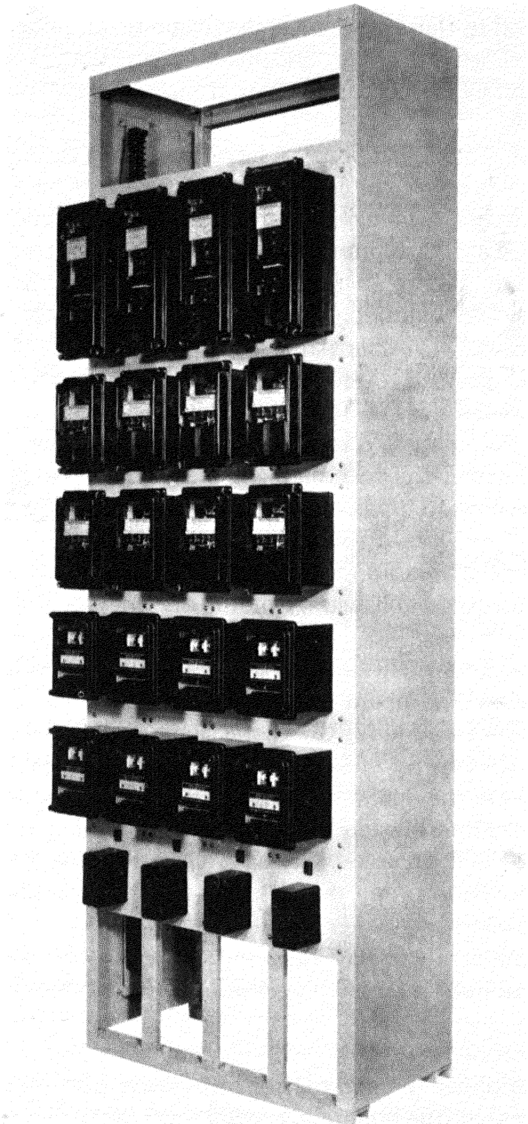

FIG. 12-9. Main coolant pump relay panel.

tection system, static components are used wherever possible. The basic component used in these systems is the magnetic amplifier; all three modes of operation—linear, bistable, and bistate—are used. No vacuum tubes are used in control circuits and relays are used only as output elements or in test circuits.

Commercial control components have been used throughout the reactor plant except when special features were required as described in preceding paragraphs. Among the commercial components used are circuit breaker control switches, auxiliary relays, motor control linestarters, and circuit breakers. Figure 12-9, a view of the relay rack for the protection circuits of the four main coolant pumps, shows examples of the types of components used. On this panel are mounted instantaneous and time delay overload protective relays, phase unbalance relays, and the power relays used to sense loss of flow.

Motor control centers contain the linestarters used for energizing all of the AC motors in the reactor plant except rod power motor-generator sets and main coolant pump motors, which are energized directly by circuit breakers.

The main coolant pump motors require winding reconnection for speed changing; this is accomplished by motor-driven 2400-volt switches.

12-1.7 Radiation monitoring. Radiation monitoring falls into two categories, safety and operational. The safety radiation monitoring system provides information for the protection of personnel in and around the plant, while the operational radiation monitoring system provides information used in the operation of the plant. The nuclear instrumentation system provides information for control of the reactor itself.

Alpha, beta, and gamma rays, as well as neutrons, are monitored by Geiger tubes, BF_3 proportional counters, scintillation counters, and ionization chambers.

Although the arrangement varies according to the specific application, each item of radiation monitoring equipment contains the following basic components:

(1) A detector.
(2) A preamplifier (if required) or impedance matching network.
(3) Signal cable (for remotely located detectors).
(4) A read-out panel.
(5) A power supply.

Five self-contained mobile trailer safety monitoring stations are used for monitoring any location on or off the site. A number of hand and foot monitors are used at various locations throughout the plant. The radiation monitoring equipment is practically all conventional; commercial equipment is used wherever possible.

12-2. Mechanical Components*

Mechanical components that handle water circulating through the core of a pressurized water reactor plant require special design treatment because of two basic requirements not normally associated with equipment available for industrial applications. *First, materials* must be carefully selected to insure satisfactory operation for a maximum life and to minimize the possibility of introducing undesirable corrosion products into the reactor coolant and reactor coolant auxiliary systems. *Second, leakage* to the ambient must be minimized to improve plant reliability and to protect operating personnel from airborne radioactivity. For these reasons, conventional commercial components, such as pumps and valves, were not used in the Shippingport plant to handle hot pressurized reactor coolant.

12-2.1 Materials of construction. All materials that come in contact with reactor coolant circulating through the reactor are of the highly corrosion-resistant type. As explained in Chapter 1, use of such materials is expected to minimize the build-up of radioactive particles on the surfaces of reactor plant equipment and to minimize deposition of corrosion products on heat transfer surfaces. In selecting materials, particular emphasis was placed on applications involving rotational or linear motion, such as pump bearings and pistons in hydraulically operated valves. Wear of these parts must be minimized to prevent minute particles from entering the coolant systems. Such particles form additional corrosion products and could also cause other equipment in the plant to malfunction. Certain elements, such as cobalt, become highly radioactive when exposed to nuclear radiation. Hence, it is necessary to control the quantities and locations of materials containing these elements, since their activated corrosion products in the reactor coolant could deposit on pipe and equipment and reduce plant accessibility for maintenance and repair. Rigid materials control was invoked for fabrication and construction of all mechanical equipment. Certified data were obtained on the chemical composition and mechanical properties of all materials.

While selection of materials is important, it is equally important to handle and process them properly. Therefore, close control was maintained over manufacturing procedures such as welding, fabrication, and cleaning the different forms of plate material, castings, and forgings.

To check vendor compliance with material and manufacturing specifications, all major components were closely inspected during manufacture. Particular attention was given to detection of material defects. In addition

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* By M. L. Sloman, Westinghouse Bettis Plant, and B. T. Resnick, U. S. Atomic Energy Commission.

to visual inspection for surface imperfections, fluid penetrant testing was employed for some components to detect cracks. All castings and welds were inspected by radiography where satisfactory radiographs could be obtained. Weld joint configurations which did not permit satisfactory radiography were inspected by either the liquid penetrant or the magnetic particle method.

12–2.2 General design requirements. Mechanical components that handle reactor coolant water were designed for a rating of 2500 psig; in most cases the design temperature was 650°F. However, the average operating conditions for this equipment are 2000 psig at 523°F. Notable exceptions are: (1) the pressurizer and associated equipment, where a saturation temperature of 636°F exists at 2000 psig, (2) equipment in the valve operating system and coolant charging system, with operating conditions of 3000 psig at 120°F, and (3) the hydrostatic test pump and auxiliary equipment, with operating conditions of 4500 psig at 120°F.

Mechanical equipment installed in the reactor coolant and reactor coolant auxiliary systems must be sealed to prevent radioactive coolant leaking to the containers. This requirement is accomplished either hermetically by seal welding or by gasketed closures with provisions for seal welding in case gaskets leak during plant operation. Special canopy and omega seal weld rings were developed and tested. The canopy seals have either a single or a double joint. The former is machined integrally with one of the parts to be welded, while the latter is a machined ring with proper weld preparation at both ends.

Proposed component designs were carefully reviewed from the standpoint of repairs. Since all equipment is welded into the piping systems, wherever possible the designs incorporate accessibility of all internal parts for repair or replacement without the necessity of cutting a major strength weld. It is costly and time consuming to replace a welded-in unit, since it involves weld grinding and/or cutting, repreparation of the cut ends for rewelding, and rewelding. Where possible, reactor coolant system components were designed to last the life of the plant.

To prevent introduction of impurities into the coolant systems, all system mechanical components were thoroughly cleaned and inspected to rigid requirements before installation. Freedom from extraneous material is important because, depending on its form and amount, such material can interfere with the movement of parts, accelerate corrosion and wear, contaminate the reactor coolant, and foul heat transfer surfaces.

After each component had been fabricated and tested at a vendor's plant, the unit was cleaned, dried, inspected, and then sealed with moistureproof, dustproof seals prior to shipping. Before installation, the seals were removed, and the unit again inspected and, if necessary, recleaned. Units were welded into the system in a clean area by techniques designed

to prevent dirt from entering the system. Consumable insert type welds with inert gas back-up were employed for the root passes to prevent loose weld spatter, scale, and slag from contaminating the system and to eliminate crevices which could act as stress raisers and traps for radioactive particles in the coolant.

All mechanical components were designed, built, and tested in accordance with applicable government and commercial codes. Pressure vessels were constructed by qualified vendors under the ASME Boiler and Pressure Vessel Code, Sections I or VIII, which are recognized and used in the State of Pennsylvania. For auxiliary equipment, such as pumps and valves, the applicable code design values were used and the completed equipment was hydrostatically tested to code values.

12–2.3 Performance testing. In some instances, mechanical component designs were radical departures from the conventional. To ensure product quality, it was imperative to construct prototypes before beginning manufacture of production units. Prototype equipment was exhaustively tested under simulated operating conditions. This included tests for compliance with all phases of the specifications as well as accelerated life tests for the expected number of operating cycles. For example, a prototype main coolant pump was built and thoroughly tested because the pumps were much larger than previous designs and used a higher voltage (2300-volt) motor, although many of the mechanical features had been proven on other canned motor pumps. After testing, the pump was disassembled and inspected for signs of excessive wear and defects. Any essential design modifications were factored into the production units. To permit rapid plant construction and insure maximum plant reliability, as many mechanical components as possible underwent production tests (at suitable test facilities) before installation in the plant. This increased assurance that the equipment would operate satisfactorily in the plant and should reduce to a minimum weld cutting, rewelding, and other work to install replacement units that would be required if trouble should develop.

12–2.4 Steam generators. The steam generating equipment consists of four units, each comprising a heat exchanger, a steam drum, and connecting piping. Two different types of heat exchangers, the U-tube and the straight tube, are used to evaluate relative performance of the two designs. Two U-tube units were supplied by the Babcock and Wilcox Company, and two straight tube units by Foster Wheeler Corporation.

All the steam generators are all of the natural circulation, two fluid type. In each, the steam drum is mounted above the heat exchanger, and the two units are connected by suitable risers and downcomers. Each heat exchanger is of the shell and tube type. Reactor coolant flows through the

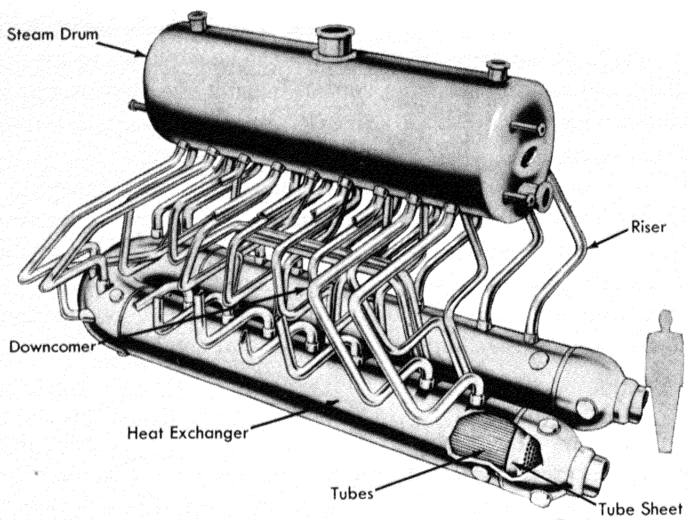

FIG. 12–10. Babcock & Wilcox steam generator.

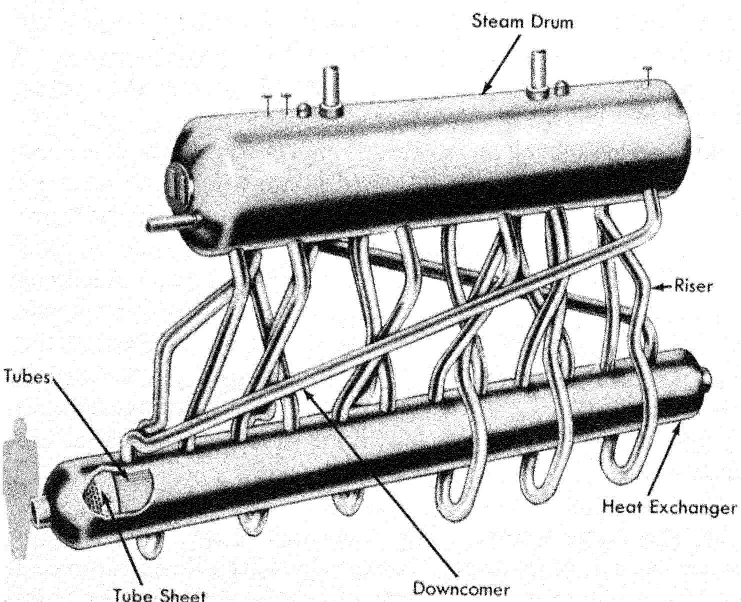

FIG. 12–11. Foster Wheeler steam generator.

tubes and steam is generated on the shell side. The steam passes upward to the steam drum through the riser tubes. Standard types of separator and scrubber elements are used in the steam drums to separate the excess water from the steam. Steam leaves the top of each steam drum through an outlet line.

The *Babcock and Wilcox steam generator* (Fig. 12–10) contains 921 stainless steel tubes $\frac{3}{4}$ in. in diameter and averaging 50 ft in length. The shell is made of carbon steel. The tube sheet on the reactor coolant side and the internal surfaces of the end heads are clad with type-304 stainless steel. Risers, downcomers, and the steam drum are made of carbon steel.

The *Foster Wheeler steam generator* (Fig. 12–11) consists of 2096 stainless steel tubes $\frac{1}{2}$ in. in diameter and $31\frac{1}{4}$ ft long. The shell is fabricated from solid stainless steel, as are the tube sheet and end heads. Risers, downcomers, and steam drum are of carbon steel.

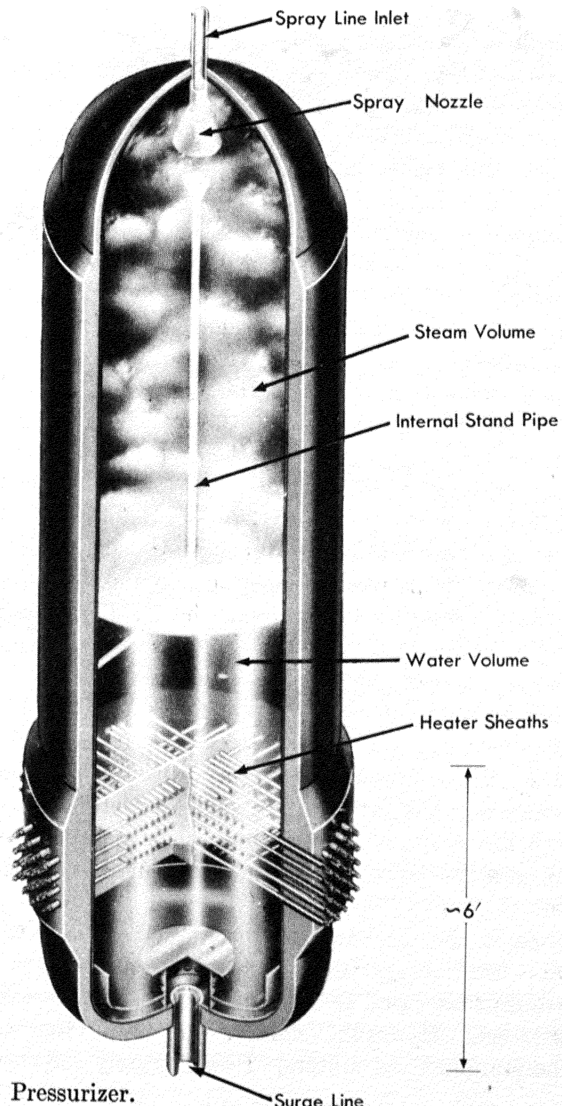

FIG. 12–12. Pressurizer.

12–2.5 Pressurizer. The pressurizer (Fig. 12–12) is a vertical pressure vessel containing steam and water. To increase pressure in the reactor coolant system, electric heaters are employed to generate steam; to decrease it, steam is condensed by a water spray system.

Because the pressurizer is so large (approximately 18 ft high and 5 ft in diameter), considerable savings were made by constructing the pressurizer of carbon steel in lieu of stainless steel. The carbon steel shell was clad with stainless steel on the inner surfaces to provide the required corrosion resistance.

The heaters, rated at 2500 watts each, are of the metal sheath type. These removable heaters fit inside stainless steel heater wells which are an integral part of the vessel. The heater wells are arranged in four quadrants in the lower shell section; the pressurizer has 342 heaters.

An internal standpipe provides a reference pressure for the D/P cell used in determining pressurizer water level.

12–2.6 Canned motor pumps. To ensure against leaks, canned motor pumps (Fig. 12–13) are used in the reactor coolant system, the discharge and vent system, and the failed element detection and location system. These pumps are of the same basic design, with canned rotors and stators, and water lubricated bearings.

The main coolant pumps are of the vertical, single-stage, centrifugal type. The coolant circulates between the rotor and the stator, lubricating the bearings and cooling the motor. The motor is mounted directly on the pump casing, and can be seal welded to the casing, forming a leakproof unit. This Westinghouse pump is designed to deliver 18,300 gpm of coolant at 340 ft of head. The motor develops approximately 1500 hp.

Canned motor pumps used in previous applications are equipped with 440-volt motors. A 440-volt motor on the PWR pump would have had to be extremely large and would have increased the difficulty of designing cables, terminals, and switchgear. It was therefore decided that the power supply would be 2300 volts.

The pump bearings are graphite (Graphitar) and nitrided 17–4 PH stainless steel. These materials have been used in previous pumps, where long-time operation has shown them to be entirely satisfactory.

The pump has a single motor winding, with taps and connections to provide full and half speed operation. Nominal full speed of the pump is 1800 rpm.

The pumps in the discharge and vent system and the failed element detection and location system are a standard and commercially available type. The motors are rated at 15 hp maximum, and are constructed mainly of stainless steel. Hydraulic ends to suit individual requirements are fitted to the standardized motors. Power supply for these pumps is 440

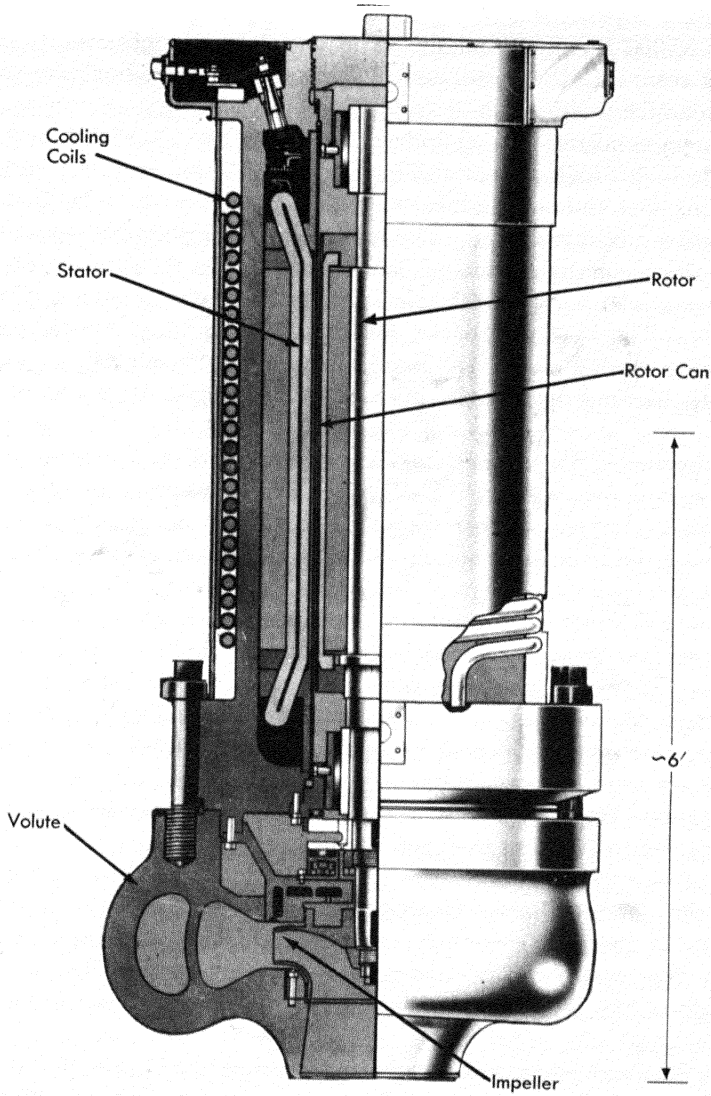

FIG. 12-13. Canned motor pump.

volts, three-phase, 60-cycle, and the units are of the vertical, single-stage, centrifugal type. Seal-weld provisions are included as in the main coolant pumps.

12-2.7 Purification components. *Demineralizers* are provided in the reactor coolant purification system to remove soluble and insoluble contaminants. The demineralizer contains 25.2 ft^3 of resin, and will accom-

modate 25,000 lb/hr of coolant. The vessel has connections to permit charging resin into the vessel and flushing the spent, radioactive resin to the radioactive waste disposal system.

The demineralizer is a cylindrical stainless steel pressure vessel approximately 6 ft high with an inside diameter of 31 in. Installed vertically, it contains inlet and outlet replaceable filter assemblies, an inlet distributor plate, and a mixed resin bed. The inlet distributor plate provides an even flow of the incoming stream of coolant across the resin bed. The resin bed is contained between the inlet distributor plate and the outlet filter assembly. The inlet and outlet filter assemblies prevent resin particles from being carried out of the demineralizer by the normal coolant flow and from backing up into the inlet line during special back flow operations.

The demineralizers are designed and constructed for a pressure of 2500 psig in accordance with the ASME code for unfired pressure vessels. The replaceable filter assemblies are sufficiently rigid to withstand a 200 psi differential pressure in either direction if the filtering units should clog completely.

The *resin* cannot withstand temperatures above about 130°F, so the reactor coolant must be cooled before it enters the demineralizer. After the water leaves the demineralizer, it must be heated before re-entering the reactor coolant system. To accomplish this cooling and subsequent reheating, a regenerative heat exchanger and a nonregenerative heat exchanger are used in a two stage process. Water from the reactor coolant system enters the tube side of the regenerative heat exchanger at 508°F and is cooled to 193°F at the exit. It then enters the tube side of the nonregenerative exchanger and exits at 120°F, the temperature at which it enters the demineralizer. Component cooling water flows through the shell side of the nonregenerative unit, entering at 100°F and exiting at 160°F. The reactor coolant discharged from the demineralizer enters the shell side of the regenerative heat exchanger at 120°F and is heated to 435°F.

Both *heat exchangers* are U-tube units primarily because of the large temperature gradients, especially in the regenerative units, to which they are subjected. Tubes, tube sheets, and shells are stainless steel, with all welded construction.

12–2.8 18-inch reactor coolant system valves. Each reactor coolant loop contains four 2500 psi, 600°F stainless steel parallel disk 18-in. stop valves, two hydraulically operated and two motor operated. These valves are used to isolate the loop from the reactor. Closest to the reactor in each end of all loops are the hydraulically operated valves. They are remotely controlled to permit isolation of a reactor coolant loop if a pipe or a piece

of pressurized equipment ruptures. They also permit isolation of a loop requiring repair while the reactor is in operation.

The *motor operated valves* are in the boiler chambers and are operated by a portable motor or hand wheel. These valves together with the hydraulic valves provide double isolation of an isolated loop from the reactor vessel.

The *hydraulically operated valves* are hermetically sealed at all times. The motor operated valves are hermetically sealed at all times except during their operation, which requires the removal of a seal-welded cap and the installation of the portable operator. Stem packing prevents excessive leakage to the ambient while the cap is removed.

Parallel disk gates are incorporated into both types of valves, which have similar flow control operating principles. (An 18-in. hydraulically operated valve is shown in Fig. 12–14.) The disks seat against flat vertical parallel seats in the body. Springs between the disks spread and force them against disk holder restraints and the tops of the body seats when the valve is in the open position. In the closed position, a pressure differential across the valve provides the seating force and seats the downstream disk only.

The hydraulic valves are operated by applying high pressure water to either the upper or lower side of the piston in the hydraulic cylinder, for closing or opening, respectively. The piston is solidly connected to the valve stem. Spring-loaded latches bear against lugs on the valve stem to maintain its position while operating pressure is balanced in the cylinder. Valve position is remotely indicated by means of magnetic steel slugs that are moved by either the disk holder at the bottom of the valve or the piston stem extension at the top. The slugs move into slender caps on either end of the valve and their motion is picked up by *E*-core position indicators mounted over the caps.

The parallel disk gate valve was selected over the wedge gate type to avoid the extreme difficulty that could develop in opening a wedge gate type valve under two different conditions: (1) When the valve is closed while hot, subsequent cooling contracts the body, causing a heavy squeeze to be applied to the gate. (2) With a hydraulically operated valve, very high pressure valve operating water may leak from the hydraulic cylinder past the valve stem, pressurize the valve body volume above the wedge gate and drive the gate too tightly into its seats. One disadvantage of the parallel disk design, however, is the continuous dragging contact between the disks and the body seats during operation.

The bodies of the 18-in. stop valves are the largest type-304 stainless steel valve castings ever made, and required extensive inspection by radiographic and dye check methods. A prototype motor operated valve was manufactured and thoroughly tested. It was this satisfactory test

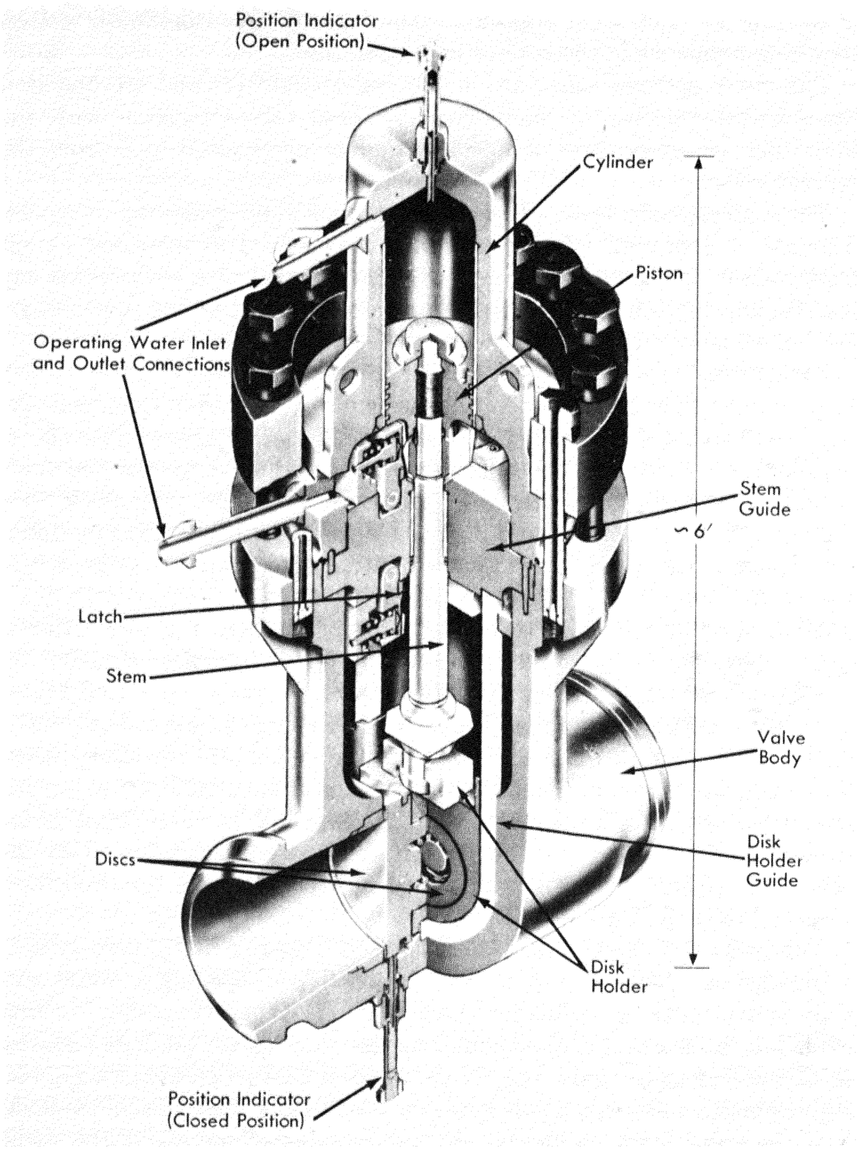

FIG. 12–14. Hydraulically operated valve.

experience that led to the use of castings instead of forgings for all 18-in. reactor coolant valves, including the check valves, which are described later.

Each reactor coolant loop also contains one 18-in., 2500-psi, 600°F hermetically sealed stainless steel tilting disk check valve to prevent ex-

cessive backflow through the loop in the event of pump stoppage. This valve is designed to unseat after closing and permit low head thermal circulation in the normal direction to handle decay heat removal. In a nonisolated loop in which the main coolant pump is stopped with other pumps running, the valve will be in the closed position. Its design also permits a backflow of approximately 100 gpm to provide means for maintaining temperature in the loop. This may prove to be an operating convenience and also helps ensure against a "cold-water accident" if the pump is accidentally started (see Chapter 11). Backflow is through a small hole in the disk.

The valve disk is supported by two pinions extending inward from either side of the body. The centerline of the pinions is behind the disk and above the flow path centerline so that the disk, when opening, swings into a horizontal position in the flow path. With no circulating head available in either direction, the disk takes a slightly tilted position away from the vertical body seat. Among advantages of the tilted disk design are: a small compact valve, low moment of inertia of the disk about the axis of rotation, and faster closing because of shorter disk travel.

12–2.9 Special valves for coolant system service. Many of the manual valves for reactor coolant and reactor coolant auxiliary system service are of special designs which include caps for enveloping all parts extending from the bonnet and for hermetically sealing the valves at all times except when opening or closing. The hermetical sealing of the cap to the bonnet, while a special feature in itself, makes the use of special noncorrosive bearing materials necessary. All moving parts are sealed inside the valve, and even though there is stem packing to prevent excessive leakage to the ambient while the cap is removed, it must be assumed that the reactor coolant will eventually leak past the stem packing and envelop all moving parts as it builds up pressure in the cap. In addition to the corrosion problem, no effective lubricant is compatible with the system water chemistry requirements. The stem threads must therefore operate immersed in water at high loads, because of the high system pressure, and without effective lubrication. For this severe service, acme stem threads cut in stellite numbers 6 and 12 operate in stellite number 6 or 17–4 PH stainless steel bonnet threads. Generally, these hard surfaces are overlaid on Haynes number 25 stems and inlaid in 304 stainless steel bonnet bores or consist of 17–4 PH solid bushings.

All *manual capped valves* are of the globe or angle type and have stellited main and back seats. Exceptional tightness is required of all seats; leakage past the disks must be not more than a very few cc of water per hour with a pressure differential equal to system pressure across the seat. Valves for isolation service are installed so that high pressure fluid is contained

under the seat to provide additional protection to the packing when the valve is closed.

Certain manual valves which handle radioactive reactor coolant are designed so that they can be operated during plant operation. This is accomplished by using an extended valve bonnet. The valve body and connected piping are in an inaccessible area while the extended bonnet projects through the shield wall into a personnel access area. This makes it possible to use less expensive manual valves where remote operated valves might otherwise be required. It also permits some maintenance of the valve while the plant is operating.

The caps are sealed with "self-energized" stainless steel O-rings, which consist of a length of small diameter tubing formed into a torus with ends butted and resistance welded together. The O-ring is installed in a special groove in the bonnet and the cap is tightened down on it. Several very small radial holes inside the O-ring allow system pressure to fill the ring—hence the term "self-energized." The top and bottom surfaces of the O-ring seal against a cap sealing surface and the bottom of the O-ring groove. The outside wall of the groove serves only as a mechanical backup for the O-ring to prevent it from bursting outwards. Practically zero leakage past the O-rings has been encountered; however, the caps are machined to permit seal-welding if excessive leakage does develop.

A threaded vent plug that seats in a conical seat in the cap provides for releasing cap pressure prior to removal of the cap. This removes the high load on the cap threads and facilitates cap removal.

For each operation of a capped manual valve, the cap must first be removed; after the valve is operated, packing tightness must be checked, and the cap resealed with a new O-ring.

There are *four types of hydraulically operated valves* in various reactor coolant and reactor coolant auxiliary systems. These include the 18-in. main stop valves, the 6-in. pressurizer surge line stop valve, 2-in. reverse seating globe valves, and 1½-in. angle valves. As the 18-in. valves were discussed above and as the 6-in. valve is similar to them, the following will pertain to the 2-in. and 1½-in. valves only.

The two-inch valve (Fig. 12–15) has a globe type reverse seating main disk, held in the closed position by a spring and by system pressure. Reverse seating merely means that the system pressure operates on the under side of the disk, pushing it up into its seat. Directly above the main disk is the hydraulic cylinder, in which an actuating piston operates. This piston is in the form of a double-ended globe valve type disk, with seats on both ends and a push rod extending from the lower end. It is closed by spring loading against its upper seat. Valve operating pressure (3000 psi) is introduced above the piston by remote operation of a three-way selector

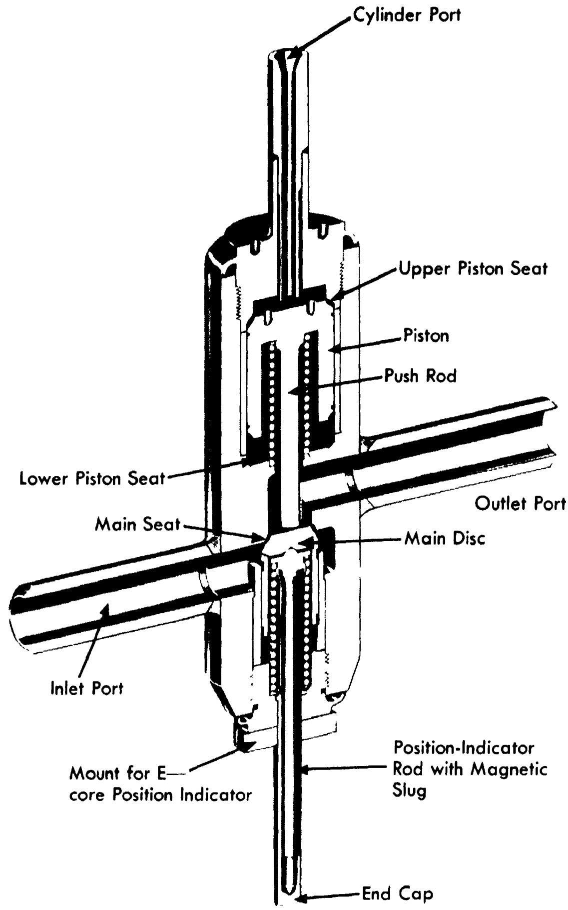

FIG. 12-15. Two-inch cylinder-operated reverse seating globe valve.

valve. This pressure forces the piston down from its upper seat into its lower seat, and at the same time the push rod moves the main disk into an open position. Venting of valve operating pressure through the three-way selector valves allows the main disk and piston to return to their closed positions by spring force. Valve position is remotely indicated by means of a magnetic steel slug that moves with the motion of the main disk in and out of a small cylinder end cap, outside of which the motion is picked up by an E-core position indicator mounted on the valve. The upper and lower piston seats are for preventing excess leakage of radioactive water into the valve operating system and for preventing excessive leakage of valve operating water into the discharge system during long periods when the valve must remain in the open position to drain a loop

The $1\frac{1}{2}$-inch angle type hydraulically operated valves are of a conventional globe type design. Connected integrally with the disk is a piston that operates in a stellite cylinder liner. The valve is operated by introducing high pressure valve operating water under the piston to open the valve or over the piston to close the valve, while at the same time water on the other side of the piston exhausts to downstream system pressure.

Three types of check valves, in sizes varying from $\frac{1}{2}$ through 4 in., are used in various coolant systems. These are lift, swing, and excess flow check types. All are of conventional design except for hermetic sealing and extensive use of stellites for seats and liners. All moving parts are accessible for repair without removing the valves from the line, except for the $\frac{1}{2}$-in. excess flow check valves. Since the only moving parts are the lifting disk, the swinging clapper disk, or the spring loaded sliding disk of the excess flow valve, maintenance should rarely be required.

Two types of relief valves are in service in the coolant systems: electrically operated and self-actuated types. The *electrically operated relief valve* is on the pressurizer and is the lowest set pressure relief valve in the reactor coolant systems; this valve will therefore be the relief valve most frequently operated. The electric valve operates from an independent pressure source and does not rely upon a balance of the lifting and seating forces as do self-actuated relief valves. This valve is operated by an external electrical signal from a pressure sensing device. The electrical signal energizes an externally mounted solenoid which drives a plunger against a fulcrum lever. The lever strikes the pilot push rod, which in turn strikes the pilot disk, causing the disk to be unseated. As the pilot disk is unseated, a hydraulic unbalance of forces is created across the main disk, causing the main disk to lift. A bellows is incorporated in the design to form a hermetic seal between the valve stem and the bonnet.

The *self-actuated relief valves* (Fig. 12–16) are of conventional design; however, they contain refinements in material selections, machining tolerances, and configuration of internal parts. Further, this particular design incorporates one unique feature which is necessary in the fulfillment of the general design requirements of the plant. This is a welded bellows which serves as a hermetic seal between the moving internal parts of the valve and the external adjusting device, and as a load-balancing member, permitting operation of the valve to be unaffected by relatively high initial and developed back pressure (downstream pressures). In addition to hermetic sealing of the valve proper, the valve bonnet, which is a gasketed closure, has a telltale nipple connection connected to a pressure-sensing device to detect bellows failure.

Air, solenoid, and motor valve operators are remotely controlled from the control room. All have manual overrides. Although all the hydraulically operated valves are indirectly remotely controlled, their operation results

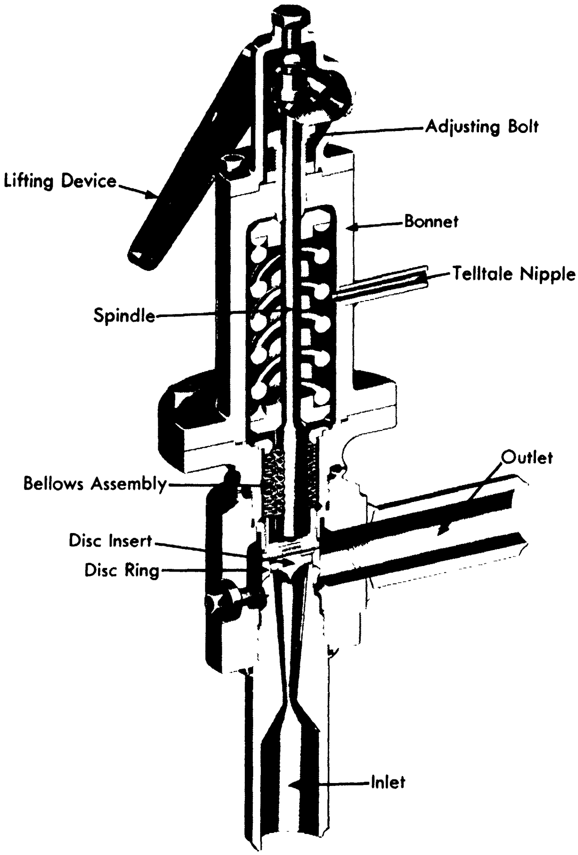

FIG. 12-16. Three-inch self-actuated reactor vessel relief valve.

from the operation of three-way motor operated selector valves which are in turn controlled by switches in the control room.

The *three-way motor operated selector valves* utilize stellited shear ring seals which seal on the very flat surfaces of a thick stellited cylindrical disk. The disk is keyed to a stem that passes through its center perpendicularly to the top and bottom flat parallel seating surfaces. Through the disk and parallel to the stem hole are two interconnected holes, which with the disk rotated to either extreme, align with the center port and one of the other two ports entering vertically from the bottom. In a center disk position, none of the ports is connected and flow is blocked. Operation is accomplished by turning a control room switch to the desired valve position. This energizes the motor in the proper rotation direction and connects the proper electrical circuits that control travel by means of microswitches mounted on the valves.

Air-operated valves of two types are used for remotely stopping or starting flow in reactor coolant systems. Small solenoid valves in the service air lines control the flow of air to the piston or diaphragm.

One type is a two-way selector valve (similar in design to the motor operated three-way selector valve except for having only two ports and a blocked position) which embodies a spring-loaded piston that must be pressurized with air to open and hold open the valve. The other type is a standard spring-closed, air diaphragm reverse seating globe valve. These valves are of a standard commercial type except for stellite trim and stainless steel bodies.

One $1\frac{1}{2}$*-in. solenoid pilot operated reverse seating stop valve* is employed in the spray regulating line to the pressurizer. A pressure sensing device in the pressurizer sends an electrical signal that energizes the solenoid, which lifts a small stem with a stellited conical end off a seat, thereby introducing upstream pressure to the top side of a small spring loaded piston. The pressure differential across the piston causes it and its integral main reverse seating disk to open, permitting the flow of coolant to the pressurizer spray nozzles.

The stem reseats when the solenoid is de-energized. A small bleed-off hole through the piston to the downstream side of the valve slowly balances the piston and eliminates the opening force. The main disk is closed by the spring of the spring-loaded piston. The stem magnetic slug is inside the hermetic seal; the solenoid coil, outside.

The *hydraulic pilot valve* (Fig. 12–17), by virtue of its hermetically sealed features, automatically directs possibly radioactive discharge water from a hydraulically operated valve to the proper discharge line without permitting any leakage to the ambient. The two lower ports are inlets from the valve operating system. Only one of these introduces high pressure at a time, by means of an upstream three-way selector valve. The pressure so introduced holds the lower globe disk closed in the manner of a check valve, while at the same time it opens the upper check valve disk, permitting flow out of the upper port to one side of the piston in the cylinder of a hydraulically operated valve. Also, at the same time, high pressure flows downward from the inlet chamber through a drilled passage to the underside of the piston under the similar set of valve disks that connect with the other inlet port. Pressure under the piston overcomes the closing spring and forces the other lower globe disk off its seat. A push rod extending from this disk transmits the force to the upper check, forcing it open. This permits flow from the low pressure side of the cylinder of the hydraulically operated valve to enter the hydraulic pilot valve at this other top port, to flow past two check type disks, and to exhaust through the low system pressure side port. Placing the three-way selector valve in the blocked position causes all disks to close, either by spring or check-

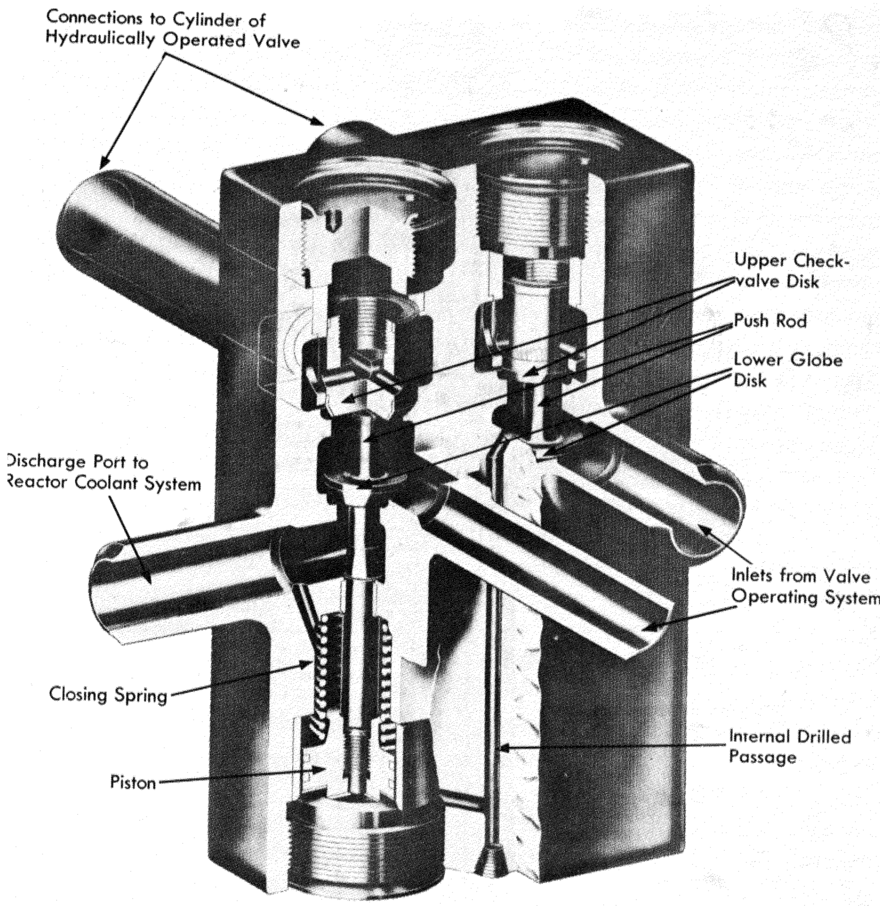

FIG. 12-17. Two-inch hydraulic pilot valve.

valve action. Pressurization of this other inlet by means of the three-way selector valve would cause similar action of the hydraulic pilot valve by merely interchanging the functions of the two identical sets of disks and pistons. Tightness of the three-way selector valve prevents backflow of radioactive water into the valve operating water supply.

To detect failure of any of the blanket assemblies in the reactor, flow from the downstream end of the blanket assemblies is sampled and monitored. To make a total of 113 individual penetrations in the reactor vessel wall would be impractical; therefore, a *multiport sampling valve* was designed to translate fluid flow from downstream of any one of the blanket assemblies individually to a monitor through only one reactor vessel penetration.

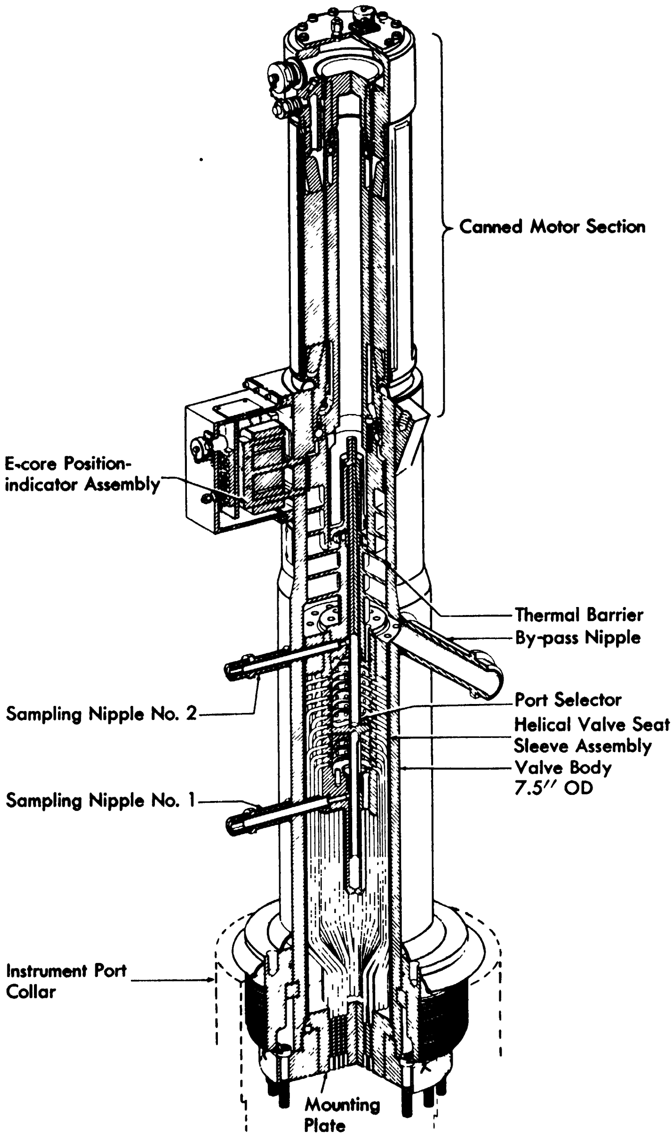

FIG. 12–18. Multiport sampling valve.

The method devised translates the flow from downstream of the blanket assemblies through 113 individual tubes to a tube sheet at the reactor vessel penetration. Flow through the valve is translated through the mating valve mounting plate containing 113 flow tubes to 113 ports in a cylinder, as shown in Fig. 12–18. Ports are spaced 30 degrees apart,

centered between a double helical thread of 1-in. right-hand lead. A port selector driven by a canned rotor motor moves within the machined helical thread and contains two pickup ports located diametrically opposite each other; therefore the valve samples two flows simultaneously. The flow path from the ports in the cylinder is split into two sections through the port selector; one flow path is from the port in one helix, through the port selector to its center, down through the port selector shaft, reversed through an annulus around the port selector shaft, and out through a sample line. The other flow path is from a port in the opposite helix, through the port selector to its center, up through the port selector shaft, out through two drilled holes in the shaft, reversing down through an annulus around the port selector shaft, and out through a second sample line. In addition, a bypass line is installed to permit continuous flow of the unanalyzed samples and to maintain a differential pressure between the sampled port flow and the cylinder chamber flow. A thermal barrier is incorporated into the design to maintain reduced temperatures at the canned rotor motor and at the position indicating device.

The valve is operated and controlled by a control rod drive mechanism and an E-core synchro-transmitter. A rotor flange containing 12 magnetic bars equally spaced around the periphery is connected to the control rod drive mechanism rotor. As the rotor operates, the magnetic bars pass magnetic slugs in the valve body; this introduces a variable differential voltage output from the E-core transmitter which is translated into a signal to start and stop the inverter operating the control rod drive mechanism.

The E-core synchro-transmitter and the control rod drive mechanism, readily available and of proven designs, were utilized in the valve design because of the short required delivery time.

The basic design philosophy was to make the valve compact, with the least number of parts, and with a low slenderness ratio. Several parts perform more than one function: The helical valve seat which contains the 113 ports in the double helical arrangement also contains the double helical thread providing the translational movement. The rotor flange, which is the driving end of the spline, contains the 12 magnetic bars for positioning control and also has several machined fins which act as a partial thermal barrier.

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CHAPTER 13

SHIELDING

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SHIELDING*

13-1. INTRODUCTION

The radiation shield for the Shippingport plant was designed to protect workers against radiation from the reactor core, the reactor coolant, and other sources.

The structures comprising the shielding may be divided into four classes: (1) the neutron (or primary) shield, (2) the internal shield, (3) the external shield, and (4) the fuel-handling shield. All four are described in detail later in this chapter.

The neutron shield is a water-filled annular tank surrounding the reactor vessel; it supplements the neutron and gamma shielding provided by the vessel wall and the water and thermal shields within the vessel. The internal shield consists of a number of walls and slabs distributed throughout the plant to shield local areas for personnel access. The external shield is the concrete enclosure housing the primary equipment in the plant. (The steel plant container, though having the principal function of preventing the escape of radioactive fluids accidentally released, also contributes to the external shielding.) The fuel-handling shield consists largely of the water in the fuel-handling canal within which irradiated fuel is handled and stored.

The need for special provisions for external shielding were minimized by locating the plant largely underground. For economy, water and ordinary construction materials (steel and concrete) were used throughout the plant for shielding, with a few minor exceptions. Wherever possible the design utilized structural elements for shielding purposes as well. For example, the cylindrical reactor vessel support is also the inner wall of the neutron shield tank. The size of the plant dictated fairly heavy concrete walls and slabs for the external shield, but the thickness was governed in most cases by structural rather than shielding requirements.

Possible economies of using heavy aggregate concrete for shielding were investigated—particularly for the heavy shield walls within the plant container where a reduction in displaced volume would be reflected in a smaller, less expensive container. Cost studies, however, indicated that savings from the reduced volume of shielding fell far short of compensating for the much higher cost for heavy aggregate concrete.

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Steel, where used specifically for shielding, is employed in a relatively minor role—it is used for the deck and railing of the access walkway around the reactor vessel closure and also for collars to prevent streaming through various pipe openings in concrete walls. However, it also serves as an important secondary shielding medium, as in the case of the walls of the reactor vessel, the reactor coolant loop piping, and the plant container, where its presence was considered in minimizing shielding requirements.

Lead has minor shielding applications, such as in the doors in the purification demineralizer enclosures. Lead shot is used as a temporary shield by placing it on a protective cover over the ion exchanger in the waste disposal system should its removal be necessary.

The shielding considerations in this chapter apply only to the reactor portion of the plant. However, the waste disposal system (see Chapter 10), because it handles radioactive wastes originating in the reactor coolant system, contains numerous vessels and piping that constitute high-radiation sources, and hence must be shielded. This shielding was generally accomplished by locating piping and storage tanks underground, but where access was needed for valves and instruments, appropriate concrete shielding was provided.

13–2. Basis for the Design

13–2.1 Plant design data. The principal plant design and operating data affecting radiation, and hence shielding design, are given in Chapter 1 and in Table 13–1.

Shutdown is assumed to be preceded by 600 hr of full power operation, defined as 60 megawatts net electrical output.

Table 13–1

Mechanical Design Data

Height of core, feet	6
Mean diameter of core, feet	6.8
U^{235} loading, pounds	165
U^{238} loading, tons	14
Zirconium loading, tons	13.8
Reactor coolant volume, cubic feet	2500

13–2.2 Radiation sources. For design purposes it was assumed that the various sources of radiation requiring shielding would have the maximum values discussed below.

A field of fast and thermal neutrons exists outside the core. Values of the neutron flux plotted radially from the face and corner of the core and through the neutron shield tank are shown in Figs. 13–1 and 13–2. Gamma

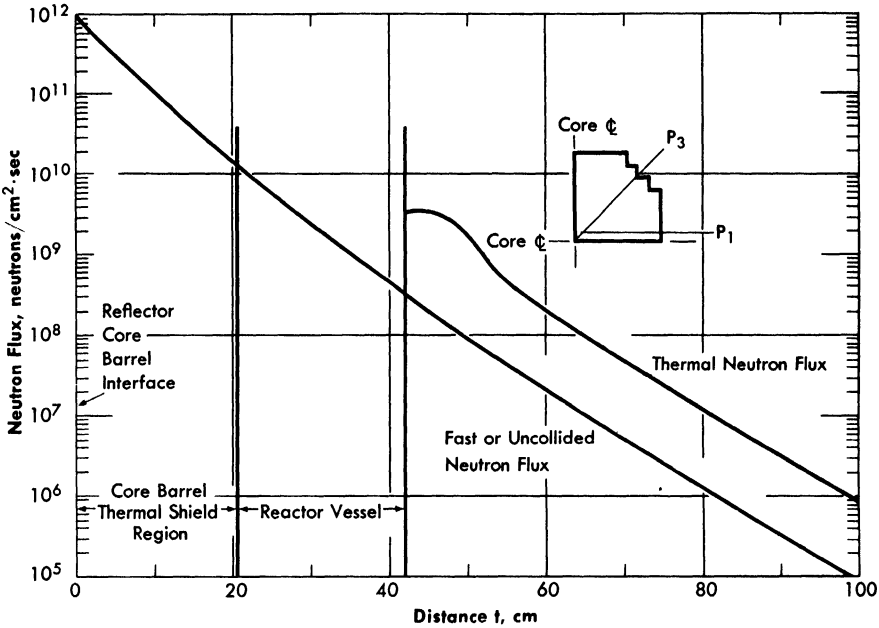

FIG. 13-1. Fast and thermal neutron distributions outside the PWR core; distance t measured along P_1.

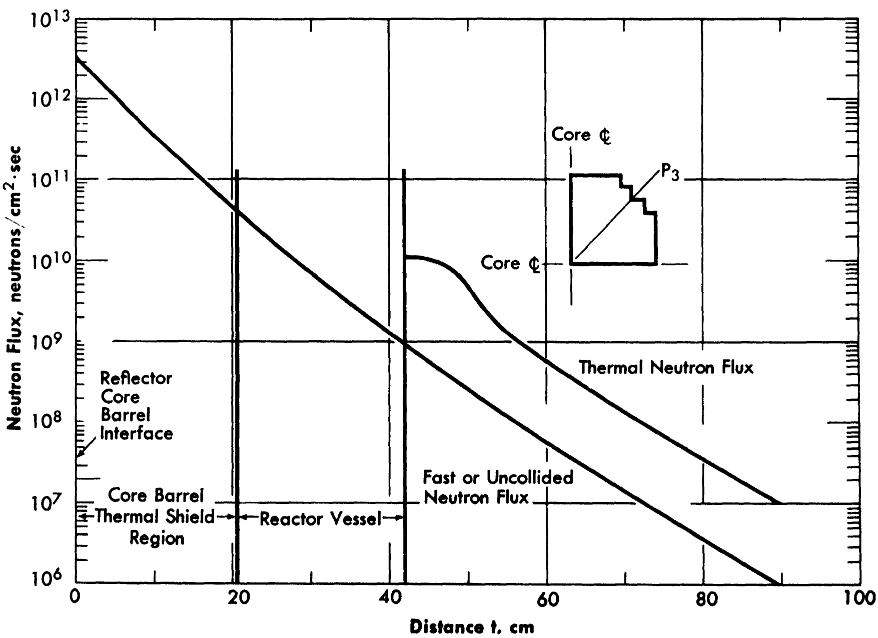

FIG. 13-2. Fast and thermal neutron distributions outside the PWR core; distance t measured along P_3.

activity at the outer surface of the reactor vessel wall produced by capture of these neutrons is calculated to be 3.5×10^9 7-Mev gammas/cm²·sec.

Maximum design values for reactor coolant specific activities are as follows:

N^{16} activity: 6×10^6 6.13-Mev gammas/cm³·sec [produced by O^{16}(n,p) reaction].

N^{17} activity: 2×10^3 1-Mev neutrons/cm³·sec [produced by O^{17}(n,p) reaction].

Corrosion product activity:* 1×10^3 disintegrations/cm³·sec (conservatively based on experience with other pressurized water reactor plants).

Fission product activity:* 1×10^7 disintegrations/cm³·sec (based on the conservative assumption of 1000 failed fuel elements).

On the basis of the above concentration in the water and of previous experience with deposition, it was assumed that the level of the fission product activity deposited on the inside surface of the reactor vessel and the coolant loop piping will not exceed a maximum of 1×10^8 1-Mev gammas/cm²·sec. The assumed maximum gamma radiation dose rate resulting from the fission product activity deposited on the inside surface of the reactor coolant piping will not exceed 200 mr (milliroentgens)† per hour one foot from the pipe.

Individual seed fuel elements removed from the core are assumed to have had a power density of 675 watts/cm³ during operation and an active length of about 6 ft. Unused fuel elements are not considered a radiation hazard.

The resin bed of the purification demineralizers may reach a maximum specific activity of 1 curie/cm³.

13–2.3 Shielding criteria. The shield was designed with sufficient margin to be adequate for the radiation generated by a possible future core of 349 Mw capacity (corresponding to 100,000 kw of electricity).

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* See Chapter 7.

† The units of radiation are the roentgen, the rep, and the rem. The *roentgen* is defined as the radiation dose that will release 83 ergs of energy in a gram of air. Since this effect results from ionization, the roentgen is applicable only to gamma radiation. The *rep* is that amount of radiation which liberates 93 ergs of energy per gram of tissue and may be applied to any type of radiation. This is approximately the release caused by one roentgen of gamma radiation. The *rem* is that amount of ionizing radiation which produces the same potential biological damage as one roentgen of gamma radiation. The RBE (relative biological effectiveness) of any type of radiation may be defined as the ratio of the physical dose of gamma radiation to the dose of radiation under consideration which will produce the same biological effects.

The types of radiation that must be considered in reactor shielding design are gamma rays and neutrons. Satisfactory shielding for these two types assures negligible levels of such other radiations as alpha and beta particles. A satisfactory gamma shield generally produces a negligible neutron level for a plant such as Shippingport, and the shielding dose rate will therefore be discussed principally in terms of mr per hour.

The shield design is based upon the criteria established by the United States National Committee on Radiation Protection and Measurement (NCRP) in 1949, amplified and discussed in National Bureau of Standards Handbook 59. The levels herein used were confirmed in later recommendations of the NCRP as published in the NBS Technical News Bulletin dated February 1957. These recommendations have now been adopted for general use by the Atomic Energy Commission. The maximum continuous dose rate for a person exposed to any radiation as thus established is 0.3 rem per week, with temporary higher rates permitted provided the dose in any seven-day period does not exceed 0.9 rem and provided further that the total dose accumulated over a 13-week period does not exceed three rems. The NCRP further recommended that no person should accumulate more than 5 roentgens for every year of his life beyond age 18.

The external shielding of the Shippingport reactor is designed to reduce the exposure in continuously occupied areas such as the control room to a maximum of 2 mr per hour. Such a dose rate is well below the recognized tolerances. For other areas, to be occupied only for maintenance or other occasional purposes, the dose rate is higher. The design dose rates for various areas in the plant are stated below. Radiation measurements made during operation show that actual levels are well within these design limits.

13-2.4 Methods used to compute shield thickness.* The primary function of the neutron shield is to attenuate the fast neutrons, arising from fission, that have not yet collided with an atom. Their biological effectiveness is many times that of thermal neutrons.

The fast neutron flux was calculated radially from the core through the thermal shields, reactor vessel, and neutron shield tank by performing a numerical integration over the core by means of the point kernel method described in Reference 1. The thermal neutron flux was determined by a combination of two-group diffusion theory and a scale-up of neutron attenuation experiments.

The calculation of the internal and external shields, specified in terms of thickness of concrete, was based on the assumed specific source activity (N^{16} in the water), a detailed approximation of the true geometry of the

* See References 1 and 2.

source with respect to the shield, and a calculated rate of attenuation for the shield material. The geometry of the radiation source was measured directly from the configuration of the mechanical components of the plant.

13-3. Description of Shielding

Shielding for the Shippingport reactor may for convenience be considered as divided into four parts: (1) the neutron shield, (2) the plant container internal shield, (3) the plant container external shield, and (4) the fuel-handling shielding. The purposes and descriptions of these categories of shielding and the dose rates resulting therefrom are explained below.

13-3.1 The neutron shield. The reactor core is surrounded on its sides by a water reflector (reactor coolant), $5\frac{1}{2}$ in. (total) of stainless steel thermal shields, and $8\frac{5}{8}$ in. of stainless clad carbon steel in the reactor vessel shell (see Fig. 13-1). Additional shielding is required, however, to reduce the radiation leakage to the reactor compartment and thus (1) prevent activation of the plant components in the reactor chamber, (2) reduce radiation heating and consequent dehydration of the structural concrete surrounding the reactor chamber, and (3) attenuate gamma radiation from the core activated sources in the vessel wall and the thermal shields.

It was determined that these requirements would be met if (1) the fast neutron flux were reduced to 10^4 n/cm^2·sec and (2) the gamma rays from the activated iron in the reactor vessel were attenuated to a dose rate of 200 to 300 mr/hr after shutdown, both measured directly outside the neutron shield. To accomplish this, an annular tank providing a belt of water 35 in. thick completely surrounds the vessel (and the bottom head) below the outlet nozzles. Sleeves were placed through the tank for the four bottom inlet nozzle penetrations, and a vented thermal expansion tank was provided above the shield. The inner and outer walls of the tank are 1-in. thick steel plate. The elements outside the vessel then, in sequence moving radially outward, are four inches of thermal insulation, a 1-in. steel plate, 35-in. of water, and an outer 1-in. steel plate. Figure 13-3 is a vertical cross section through the reactor and neutron shield.

The maximum thermal neutron flux at the outside surface of the neutron shield tank during full-power operation is calculated to be 5×10^3 n/cm^2·sec, and the maximum fast or uncollided flux, 5×10^2 n/cm^2·sec.

The dose rate at the outside surface of the shield tank 15 min after shutdown is calculated to be approximately 16,000 mr/hr. This is due to the sum of the delayed core gamma activity plus the thermal neutron activation of the steel in the reactor vessel shell and the inside wall of the

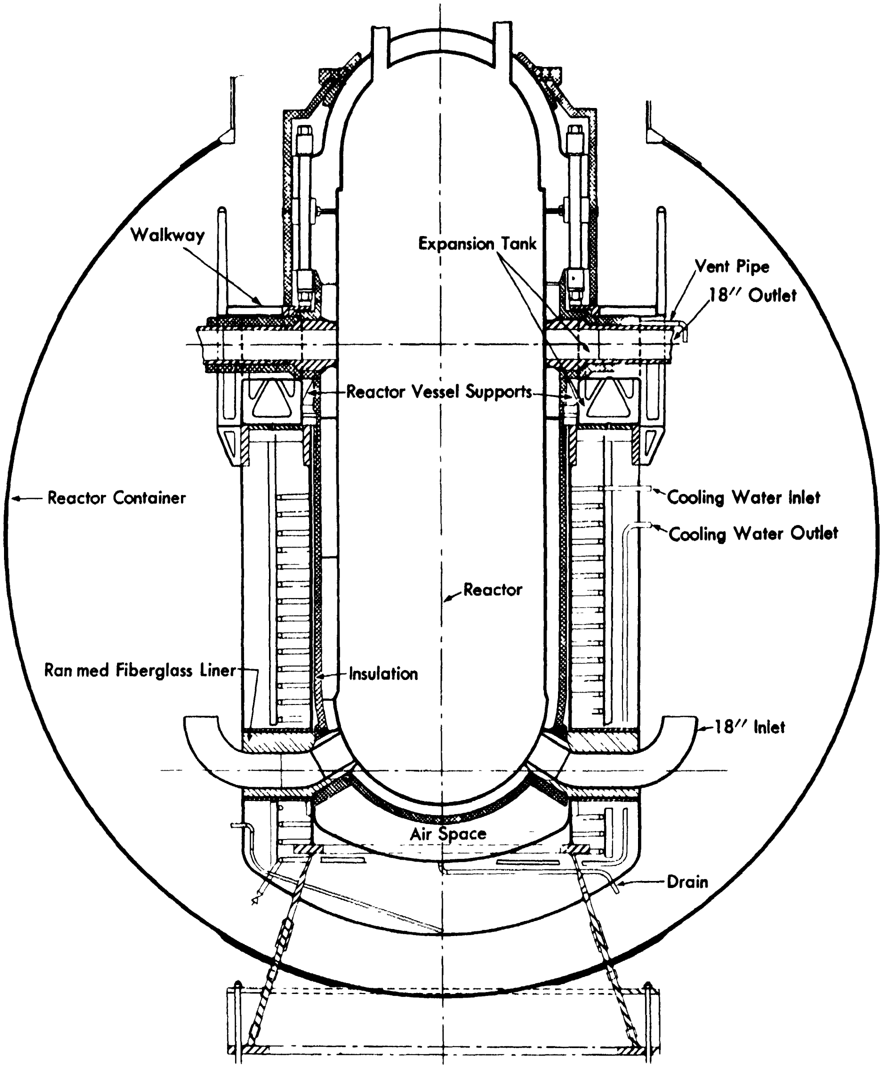

FIG. 13-3. Neutron shield tank.

neutron shield tank. Twenty hours after shutdown this dose rate has decayed to a level of 200 to 300 mr/hr. Further significant decay of the dose rate cannot be expected before approximately six months after shutdown. The accessibility permitted by these dose rates is considered satisfactory, since there is little equipment in this area which requires maintenance.

No external neutron shielding directly above the reactor is needed because of the approximately 15 ft of water within the vessel above the core.

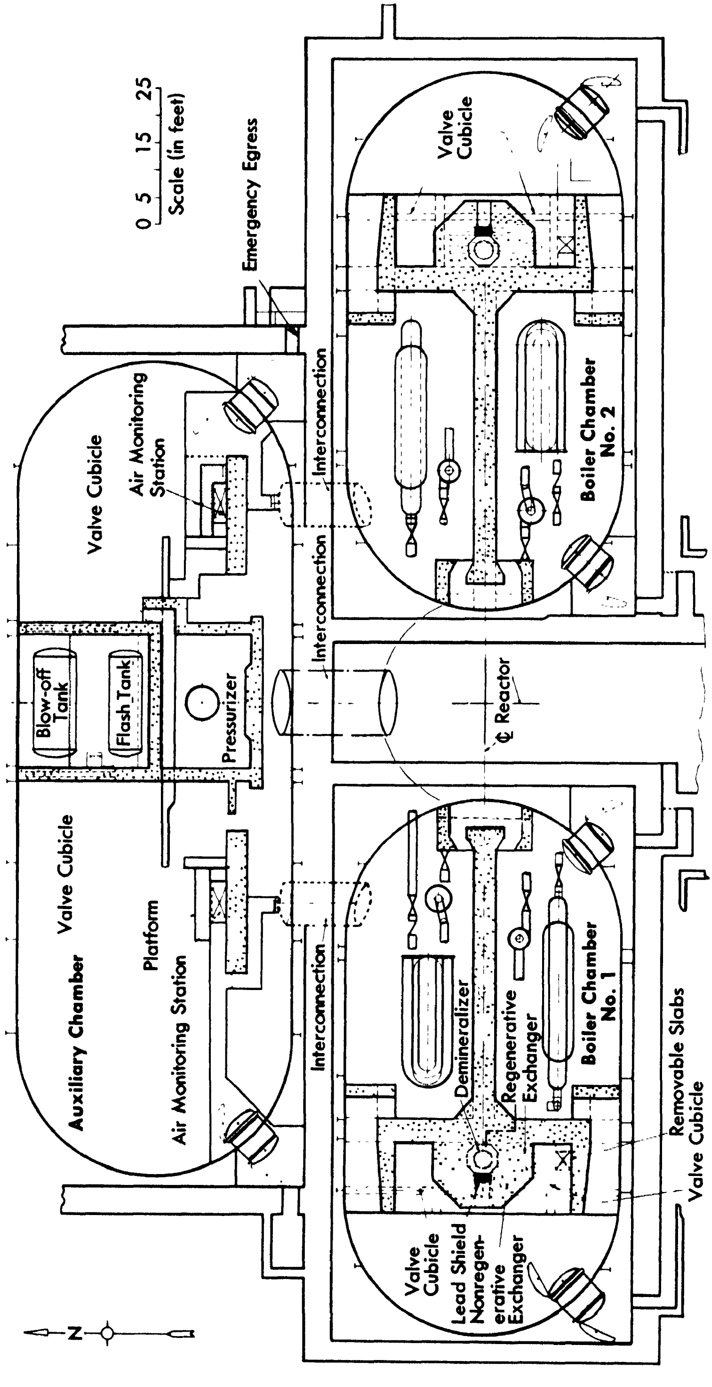

FIG. 13–4. Plan of internal shielding, boiler and auxiliary chambers.

13-3.2 The plant container internal shield. Shielding is provided inside the reactor plant container to permit access to (1) the purification valve operating cubicles in the boiler chambers, (2) the valve operating cubicles in the auxiliary chamber, and (3) an isolated boiler compartment during three-loop operation. Figure 13-4 shows details of internal shielding, which are described below.

Purification cubicles. A shielded area at the outer end of each boiler chamber permits limited personnel access during operation to a number of valves and instruments, mainly associated with the coolant purification system. These areas, called purification cubicles, are separated from the boiler loops by a concrete wall 4 ft thick, designed to limit the dose rate to 50 mr/hr at any point within the cubicles.

The large purification demineralizer vessel, which is placed in this area, is separately shielded with 6 ft of concrete, since the spent resins represent a very high radiation source. The resulting dose rate at the outside surface of this shield is 20 mr/hr from the demineralizers alone. A concrete plug in the top of the demineralizer shield can be removed for vessel replacement. A second access opening in the side of the demineralizer cell is shielded with a lead door 10 in. thick, which may be raised by twin lifting screws mounted on top of the cell.

Extensive auxiliary piping and the coolant purification system heat exchangers, both of which constitute radiation sources, are also located in this area, mainly beneath the floor. Shielding is provided by a concrete floor 1.5 ft thick, with removable sections for access to valves.

Boiler compartments. The two boiler compartments in each boiler chamber are exclusion areas during plant operation; however, internal shielding is provided inside the chambers to permit limited access to an individual boiler loop isolated from the reactor while the remaining three loops are operating at full power. This shield consists of a longitudinal concrete wall four feet thick.

The boiler chamber is separated from the reactor chamber by a concrete wall 3 ft thick. Penetrations are provided through the wall for the 44-inch diameter interconnections between the boiler and reactor chambers for passage of the coolant loop piping. The annulus between the piping and the interconnection is shielded with laminated steel collars from 4 to 6 in. thick.

The internal shielding thus described restricts to 100 mr/hr the total dose contribution from the operational sources of radiation in the adjacent reactor, boiler, and purification compartments to the isolated boiler compartment. The residual fission-product activity on the inside walls of the loop piping may contribute an additional 100 to 200 mr/hr, resulting in a total dose rate in the isolated boiler compartment as high as 200 to 300 mr/hr.

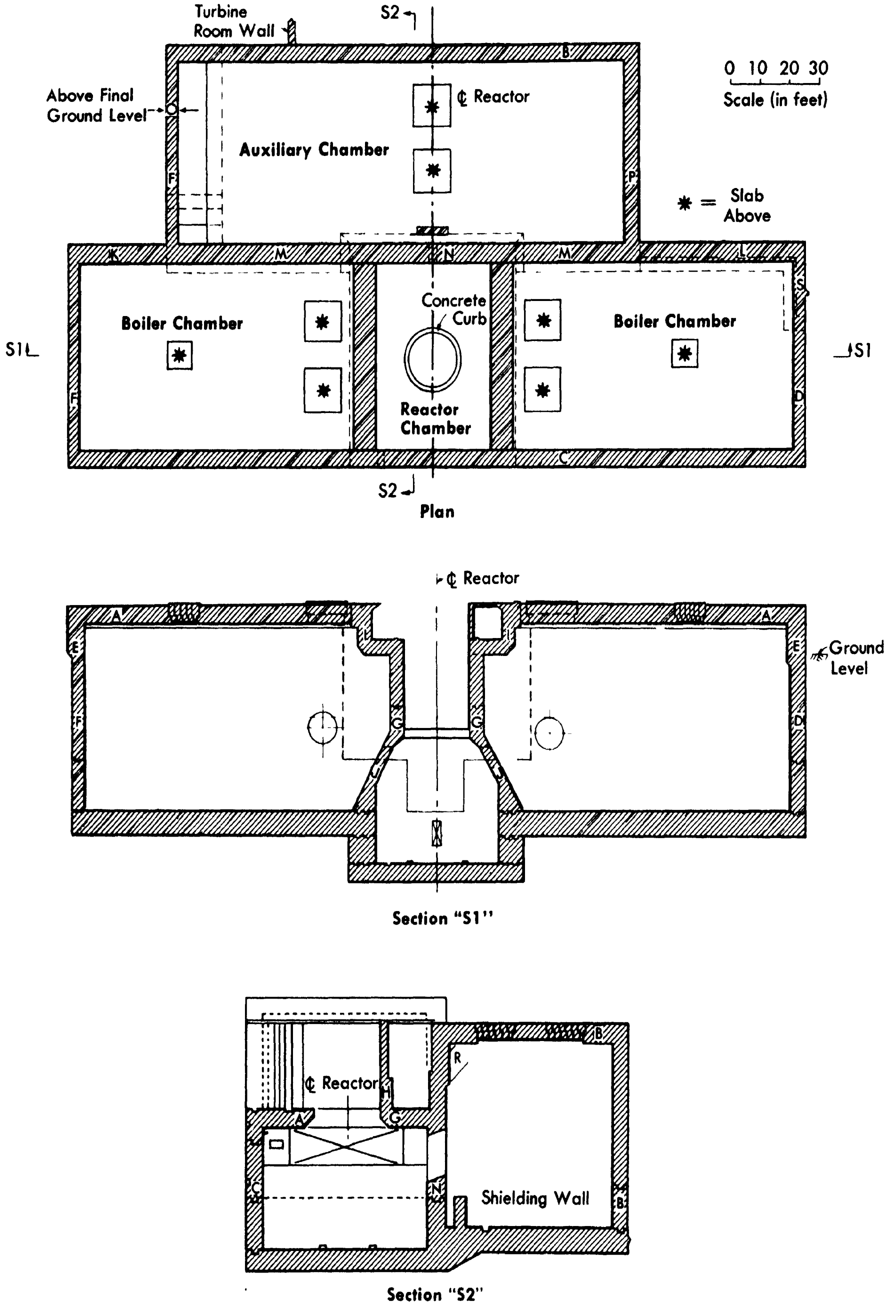

Fig. 13–5. External shield concrete enclosure.

Auxiliary chamber. Valve operating areas in the auxiliary chamber similarly require shielding from large radiation sources such as the pressurizer, the flash and blow-off tanks, and associated auxiliary piping. These areas (indicated as "valve cubicles") are shown in Fig. 13-4. The shielding within the auxiliary chamber consists of a central box-like structure with concrete walls two feet thick enclosing the above-named vessels, and two large rectangular slabs 4 ft thick placed in front of the interconnections to the boiler chambers to intercept radiation from the reactor coolant loops.

This shielding limits the dose rate in the cubicles, which are immediately outside the transverse concrete wall, to 50 mr/hr during full-power operation. The manual valves to be operated from these cubicles contain radioactive coolant and are therefore mounted inside the pressurizer and discharge tank compartments. They are operated through shielded ports in the wall. A shielded passageway beneath the compartment floor connects the two cubicles.

Some local shadow shielding is provided at points where piping penetrates the chamber and for the air-monitoring instruments, which may collect some radioactivity.

13-3.3 The plant container external shield. The external shield is a concrete enclosure, largely underground, forming the walls and roof of the structure housing the reactor plant container. Supplementing the concrete shield is the steel plate of the plant container, which is $1\frac{1}{4}$ in. thick in the cylindrical portions.

The concrete wall and slab thicknesses are shown in Table 13-2 and Fig. 13-5. Table 13-2, for purposes of comparison, lists the wall thicknesses required for shielding and for structural strength. Note that the minimum concrete thicknesses required for structural reasons are usually equal to or greater than those required for shielding.

The plant container external shield is nominally designed for a 7.5 mr/hr dose rate at the outside surface over the normally unoccupied areas, and for a maximum dose rate of 2 mr/hr at the outside surface of the shield where the surface bounds a normally occupied area such as the control room or the turbine room. Also, if a normally occupied area is bounded by two or more shield surfaces, the shield is designed to hold the radiation level to not more than 2 mr/hr. Because the surface of the top slab of the external shield is large and the distance from the surface of the source is relatively great, the resulting dose rate at any point on the outside surface of the shield may be considered constant. Consequently, the thickness requirement for the top slab of the shield is uniform throughout its area.

The density of structural grade concrete (exclusive of reinforcing steel) was assumed to be 144 lb/ft^3. The reinforcing steel, not considered in the

TABLE 13–2

EXTERIOR CONCRETE WALL AND SLAB THICKNESSES

Wall (see Fig. 13–5)	Thickness req'd for shielding	Thickness req'd for strength	Thickness used
A	4′6″	4′0″	*4′6″
		5′0″	5′0″
B	4′6″	5′0″	5′0″
C	4′6″	5′0″	5′0″
D	3′9″	3′6″	*3′9″
E	4′6″	3′6″	*4′6″
F	None	3′6″	3′6″
G	4′0″	4′0″	4′0″
		5′0″	5′0″
H	2′6″	3′0″	3′0″
I	3′9″	4′0″	4′0″
J	3′0″	3′0″	3′0″
K	3′9″	5′0″	5′0″
L	4′9″	5′0″	5′0″
M	3′3″	5′0″	5′0″
N	2′3″	5′0″	5′0″
O	4′6″	4′6″	4′6″
P	4′9″	3′6″	*4′9″
R	4′0″	4′0″	4′0″
S	3′9″	3′6″	*3′9″

* Actual thickness used was result of shielding thickness requirement.

shielding calculations because of its irregular pattern, represents a small extra safety factor.

Other areas not normally occupied, but occasionally requiring access on a full shift basis (e.g., the auxiliary power room), are shielded to a maximum of 5 mr/hr. Most of the plant container exterior walls are earth shielded, since the bulk of the plant is below ground level.

Removable concrete hatches of approximately the same thickness as the slab are provided in the top of the external shield over the hatches in the reactor plant container for removing or replacing equipment after shutdown. Penetrations through the shield for personnel access are below ground level and do not require special shielding.

Small piping, instrumentation, and electrical penetrations through the shield are arranged so that no source of radiation is directly in line with the penetration. In the case of the steam and boiler feed lines which penetrate the turbine room wall, additional shadow shields have been provided in the turbine room.

Shielding directly above the reactor is afforded by the water in the fuel-handling canal in conjunction with the steel in the reactor vessel head and the reactor container dome. The canal water level during operation provides 4.6 ft of water over the top of the mechanism dome. The steel reactor vessel head is 10.25 in. thick and the steel dome is 1.75 in. thick. The resulting full-power dose rate at the surface of the canal water directly over the mechanism dome is less than 5 mr/hr and is substantially less at the access areas alongside the canal.

13-3.4 The fuel-handling shield. The fuel-handling canal (see Fig. 4-26 for cutaway view) provides water as the primary medium for shielding irradiated fuel during removal and storage.

During handling, servicing, and storage, irradiated fuel as well as radioactive reactor components will be immersed in water, which provides both shielding and cooling.* Water in the canal is deep enough to limit the dose rate at the surface of the water above the active fuel elements to 7.5 mr/hr, except for the area adjacent to the reactor vessel head during core removal operations, and the boiler chambers, the dry pit, and the east canal stairwell during core transfer operations. Personnel will be temporarily excluded from these areas during head removal and core transfer operations. Details of the refueling procedure are described in Chapter 4.

Refueling operations introduce a number of special shielding considerations, briefly described below.

Operations on vessel head. Before refueling, it is necessary to open fuel ports, cut seal welds, remove cables, and perform various other operations on the vessel head equipment. At this time it is expected that the maximum activity of the reactor coolant will be 1×10^3 disintegrations/cm^3·sec, with the possibility of the fission products deposited on the pipe and vessel walls reaching an activity level of 1×10^8 dis/cm^2·sec. The resulting dose rate in the working area at the top of the head is about 50 mr/hr, which is considered permissible for short-time exposures. Accordingly, no special shielding was provided for this area.

Vessel head removal. To permit work on the head closure bolts and seal weld, a shielded walkway encircles the reactor vessel head above the outlet nozzles. Figure 13-3 shows a cross section of the neutron shield tank and walkway. The walkway and railing are of laminated steel plate 4.5 inches thick. The dose rate on the walkway during such operations is a function of the activity of fission products deposited on the reactor coolant pipe walls, and in mr/hr is equal to 3×10^{-6} times the residual activity in dis/cm^2·sec. For example, if the residual activity should reach 1×10^8

* An alternative method of refueling has been developed in which the fuel is removed into a shielded container. The maximum radiation at the surface of the container will be 200 mr/hr.

dis/cm^2·sec (refer to Article 13–2.2 on radiation sources) the dose rate on the walkway will be 300 mr/hr.

During the head removal operation, personnel will be excluded from the vicinity of the reactor pit while the head is being lifted, to protect them from becoming exposed to the under surface of the reactor vessel head. In addition, there will be only limited access to the area around the reactor pit. The dose rate at the top of the canal above the reactor may reach 300 mr/hr if the deposited fission-product activity on the surface of the core hold-down barrel, exposed when the head is removed, reaches 1×10^8 dis/cm^2·sec. At the same time the dose rate on the shielded walkway, because of the exposed hold-down barrel, will be approximately 7.5×10^4 mr/hr, which will render the walkway inaccessible after the head is removed. It should be noted that access to these areas is not required during refueling.

Core transfer. During the underwater transfer of a spent core to the disassembly area, personnel are excluded from the boiler containers, the stairwell along the east side of the canal, and the dry pit, because of the high radiation levels created in these areas by the core. The dose rate in the boiler compartment is approximately 800 mr/hr, and that in the stairwell and dry pit about 1600 mr/hr.

Subassembly removal. Subassemblies may be removed under water through fuel ports in the reactor vessel head. The top of the active material in the seed assemblies withdrawn in this manner is raised to within 10 ft of the surface of the canal water for the short time required to get the subassembly over the top of the port. This produces a dose rate of approximately 15 mr/hr at the canal water surface. The subassembly is then lowered to a depth of 15 ft in the transfer canal, producing a dose rate of less than 1 mr/hr at the canal water surface. A blanket subassembly removed in a similar manner produces a dose rate of 8 mr/hr at the canal water surface as it is lifted over the top of the fuel port. At a depth of 15 ft the blanket subassembly produces a negligible dose rate.

During underwater transfer of a spent subassembly to the fuel storage area, personnel are temporarily excluded from the boiler compartments, the east canal stairwell, and the dry pit. As the seed subassembly passes through the transfer canal, the dose rate in the boiler compartment reaches 100 mr/hr, in the stairwell 70 mr/hr, and in the dry pit 275 mr/hr.

The fuel storage area of the canal provides sufficient space for the subassemblies of an entire core in vertical positions under 20 ft of water. The resulting dose rate at the water surface is less than 1 mr/hr.

Each spent subassembly is loaded vertically into a shielded shipping container under water in the fuel canal. The container is used to ship the subassembly to a fuel processing facility. This loading procedure requires that the top of the active fuel material be raised to approximately 10 ft

below the canal water surface, with the subassembly in a vertical position for lowering into the container. In this position the dose rate at the water surface 30 days after shutdown should not exceed 5 mr/hr.

13–4. Other Shielding Considerations

13–4.1 Effectiveness of external shield in event of accident. For shield design purposes, the worst conceivable accident that could occur to the plant was assumed—a coolant system rupture followed by release and uniform distribution of all of the full power equilibrium core fission products throughout the reactor plant container. The permissible radiation level for such an unlikely event need only allow limited access to those areas where key controls are located. A design dose rate of 2 r/hr hour satisfies these limitations and requires approximately 5 ft of concrete. This is approximately the same thickness of concrete previously specified for normal operational shielding for the control room, auxiliary equipment room, and turbine room areas. However, subsequent studies showed that the design was very conservative—only a small percentage of the fission products would be released from the core in the event of such an accident and the radiation dose would therefore be much less than 2 r/hr.

13–4.2 Effect of future cores on radiation levels. Consideration has been given to the effect of future cores on the radiation levels in occupied areas of the Shippingport plant. The ultimate design capacity of the plant is 349 Mw of reactor heat. Since detailed design information for the future cores is not now available, the shield design was based on a scaleup of existing data to the higher power level. The specific activity used in the design of the neutron shield, the internal shielding, the external shield, and the fuel-handling canal shielding represents a value which makes allowances for the higher power core. Until detailed design information for the 349-Mw core becomes available, no computational evaluation of the existing shield can be made. However, the dose rates in occupied areas based upon the assumed source data for the future core are substantially the same as those defined in Section 13–3 above.

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Supplementary Reading

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CHAPTER 14

TURBINE-GENERATOR PLANT

CHAPTER 14

TURBINE-GENERATOR PLANT*

14-1. Introduction

The Duquesne Light Company was responsible for the design and construction of the turbogenerator portion of the Shippingport Station as part of its contractual obligation. This chapter describes the turbogenerator, related equipment in the thermal cycle, electrical generation and distribution equipment, station auxiliaries, and the structures housing these components.

14-2. Structural Design Features

14-2.1 Design flood elevation. Since the Shippingport site is near the Ohio River, which is subject to flooding, the station must be protected against flood waters which could render it inoperative and damage equipment and buildings. At the same time, economics dictated that the design flood elevation be realistic, since it affected the elevation of plant facilities and the design of portions of the buildings which are subject to hydrostatic pressure.

Elevation 706 was adopted as the design flood elevation after study of the river's flood history. The maximum flood on record occurred in March 1936, cresting at elevation 703 at the Shippingport site. Present headwater flood control reservoirs were not in operation at that time. Had they been, they could have reduced that flood to elevation 692.9. A standard design flood, developed by the Corps of Engineers, U. S. Army, could produce an unreduced flood elevation of 712.5. The present reservoirs would reduce this flood by an estimated 12 ft.

Protecting the station to the unreduced elevation of the standard design flood would be too costly, considering the remote possibility of such a flood as indicated by a flood frequency curve. However, it was considered unwise to count on the full reduction, since the amount of reduction can vary with the type of flood and the availability of reservoir storage. The adopted design flood elevation of 706 represents a partial reduction in this standard design flood. At the same time, it is consistent with the Duquesne Light Company practice of protecting their generating stations to an elevation of three feet above the 1936 flood.

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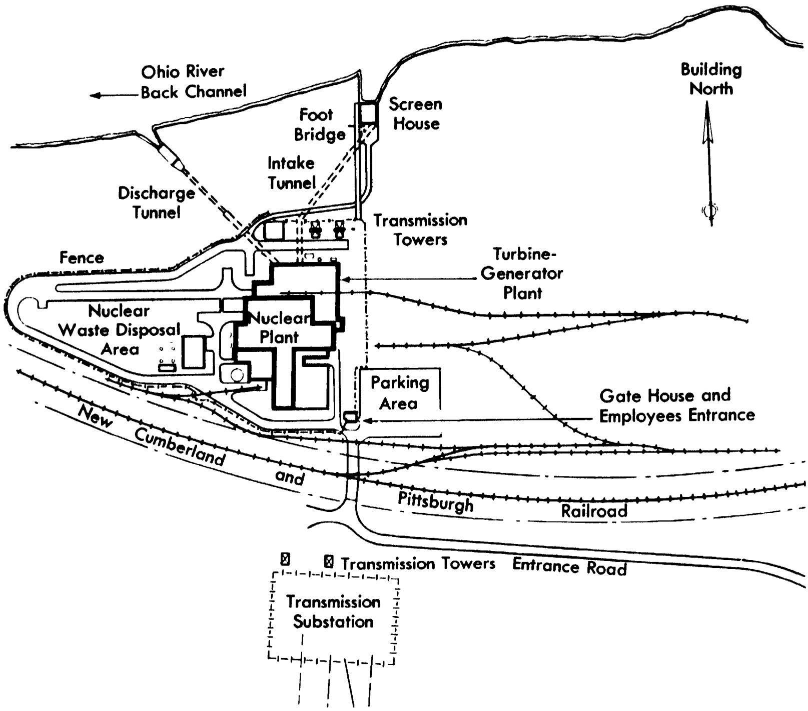

FIG. 14–1. General plant arrangement.

14–2.2 Turbine-generator building. The turbine-generator building, part of the turbine-generator plant shown in Fig. 14–1, is located north of and adjacent to the auxiliary chamber of the reactor plant container. This building supports the outdoor type turbine-generator unit and houses mechanical equipment related to it. Also contained within this building are electrical switchgear and storage space.

The silt and clay deposits in the area of the turbine-generator building (Fig. 14–2) were not suitable to support the building. It was necessary to remove the faulty material down to good sand and gravel and to backfill the area with compacted select sand and gravel up to the elevation of the bottom of the slab at elevation 673.

The building is supported on a concrete mat 10 ft thick. This thickness was determined by equating the weight of the slab and the dead load on it to the hydrostatic pressure produced with floodwater at elevation 706. The resultant slab thickness included a factor of safety against uplift and provided sufficient slab thickness to minimize deflections due to varying hydrostatic pressures. A distinct separation was provided between this slab and the reactor building to minimize difficulties that might arise because of possible differential settlements caused by different foundation

loads. Rubber water stops inserted between the slab and the reactor building maintain watertightness and allow such relative movement as might occur.

The concrete walls of the turbine-generator building were designed as vertical cantilevers to resist earth pressure (where fill was placed against the wall) and the potential hydrostatic pressure (assuming floodwater to elevation 706). Physical separation from the reactor plant walls is provided, with rubber water stops being used to exclude flood water.

The steel framing which supports the operating deck and the two floors just below is of standard beam and column design, the floor beams being pocketed in the concrete walls. A row of columns was required along the south side to maintain the separation between the turbine-generator building and the reactor plant fuel-handling building. Structural connections are made with high-strength bolts, as these can be installed more quickly than rivets. Because of the tight schedule, any saving in time was important. The railroad track at the operating floor level is supported directly on beams under the rails.

The operating deck at elevation 723 is an $8\frac{1}{2}$-in. concrete slab. To permit laydown of equipment and material, a live load of 500 lb/ft^2 was used in design. A floor drain system was installed, and a colorless silicone waterproofing material was applied to the deck. Special effort was made to seal the joints between steel and concrete, for example at the railroad track, crane rail, etc. At the isolation joint between the turbine-generator support and the deck, an underfloor gutter system intercepts leakage and conducts it to the station drainage system. The two intermediate floors are either 5-in. concrete slabs or steel grating. In general, concrete floors are used in areas where the switchgear is located, to provide a smooth rolling surface and to permit accurate positioning of the equipment. Grating is used around most of the auxiliary mechanical equipment.

The turbine-generator support is constructed of heavy box girders and columns built up with plates and angles. To keep the top surface in a flat plane, the columns were proportioned with their vertical deformations as nearly equal as possible and within the maximum allowable of 0.030 in. The maximum vertical and lateral deformation of the girders was limited to 0.020 in. The girders and columns are filled with concrete for added mass to minimize vibration. The support is erected directly on the building foundation slab at elevation 685, making use of the mass of this heavy concrete slab to reduce vibration. A complete separation is maintained between turbine-generator support and building steel and floors to reduce to a minimum the possibility of transmitting vibration to other facilities. High-strength bolts were considered for the connections of the turbine-generator support. However, it was decided to use rivets because of the lack of industrial experience, at that time, in the use of high-strength bolts in a structure subject to vibration.

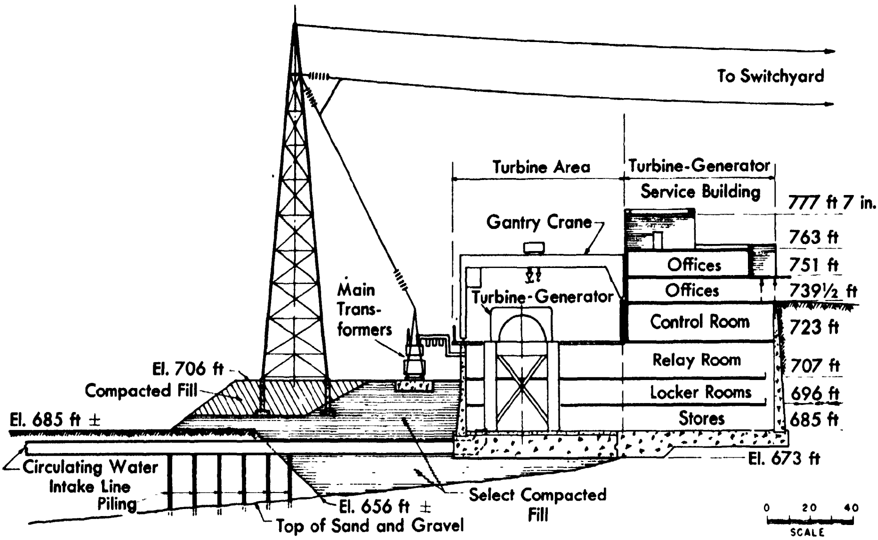

FIG. 14–2. Sectional view of turbine room, turbine-generator service building, and adjacent areas.

14–2.3 Turbine-generator service building. The turbine-generator service building is east of the auxiliary chamber of the reactor plant container and south of the turbine-generator building.

Figure 14–2 shows a cross section through the building. Contained in this building are administrative and service facilities essential to the operation of a power generating station. The control room for the nuclear and turbine-generator plants is located at elevation 723 for convenient access to the major operating equipment in both plants. By taking advantage of the terrain and locating the offices on the two top floors, access to the office was conveniently provided at the upper ground level.

A good sand and gravel stratum was present in the area of the turbine-generator building at the elevation of the bottom of the foundation. The mat type foundation slab was placed directly on this stratum. The design of this slab was similar to that described for the turbine-generator building. Separation from the nuclear plant container concrete enclosures was again provided, the only physical connection being the rubber water stop.

The building frame is of steel beam and column design, with the beams along the east wall pocketed in the concrete wall below elevation 739 ft 6 in. Above elevation 739 ft 6 in., columns along the east wall support the beams framing into that wall. A row of columns along the west side permits separation of the turbine-generator service building from the adjacent nuclear plant structures. To provide the unrestricted view required in the control room, interior columns were eliminated on that floor; the

columns above are supported on plate girders spanning the control room. Walls below grade, where required, are of reinforced concrete, designed as vertical cantilevers. Hollow concrete blocks are used for all above-grade exterior walls and interior masonry partitions. The exterior walls have been treated with a colorless silicone paint for waterproofing.

14–2.4 Screenhouse. Plan and sectional views of the screenhouse are shown in Fig. 14–3. The screenhouse functions primarily as an intake for circulating (condenser cooling) water and houses the associated pumps, screens, and other equipment. The intake sill and pump suction bells are located well below any anticipated low water level. The elevation of the top of the structure was set by the design flood elevation of 706.

This structure is erected directly on the same sand and gravel stratum that supports the other major structures. The base of this structure, a

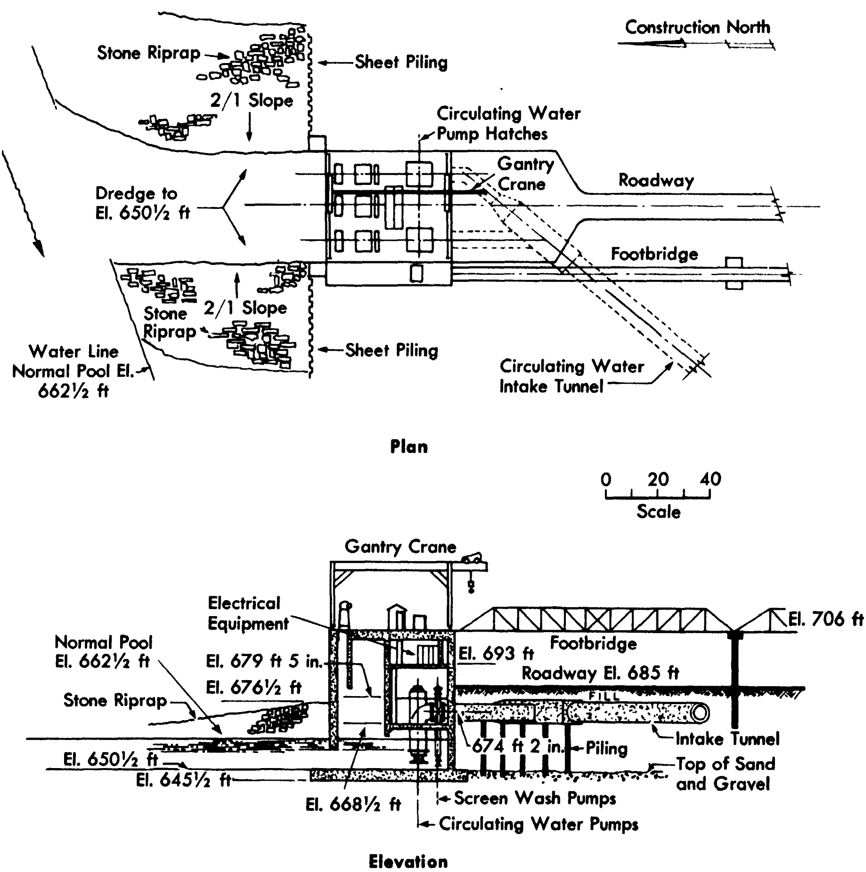

FIG. 14–3. Screenhouse plan and elevation.

concrete slab, also serves as the floor of the intake well. The cofferdam steel sheet piling, which was driven to bedrock and used as forms for the concrete wall below grade, was left in place to guard against erosion.

The screenhouse walls and floors are constructed entirely of reinforced concrete. By sealing the pump chamber and designing the floors and walls to withstand hydrostatic pressure to elevation 706, it is possible to operate with floods up to that level.

A footbridge spans the distance between the filled area north of the turbine-generator building and the screenhouse, because the low area near the screenhouse will be inundated during floods. This bridge also carries electric conduits to the screenhouse equipment. A 10-ton gantry crane handles, through the deck hatches, equipment for necessary maintenance.

14-2.5 Circulating water intake and discharge lines and outfall structure. Several different foundation conditions were encountered in designing the circulating water intake and discharge lines. To minimize differential settlement that might crack the lines, the type of construction and foundation had to be varied to meet each particular condition. The cast reinforced concrete transition piece near the screenhouse merges the two pump discharge lines into a single conduit having walls 2 ft thick with inside dimensions 6 ft 6 in. square. This transition piece is supported on steel H-piles driven into the sand and gravel stratum on which the screenhouse is erected.

At the turbine room end also, the intake conduit is a square concrete tunnel of the same dimensions. Near the turbine room the conduit is supported on the same compacted select fill as is the turbine room. North of the compacted fill the route of the tunnel passes through and over a clay layer overlaid with about 15 ft of fill. To prevent settlement due to the compression of the clay by the fill, the concrete tunnel is supported on steel H-piles driven into the sand and gravel. The rest of the line between the transition piece at the screenhouse and the square tunnel at the turbine room end is made up of an 84-in. reinforced concrete pipe. Although this pipe passes through the clay stratum, no significant settlement is expected because the pipe full of water weighs approximately the same as the clay it displaces. The circulating water discharge line is also composed of cast-in-place concrete tunnel and reinforced concrete pipe, since the foundation conditions encountered are similar to those of the intake line.

The discharge tunnel terminates at a concrete outfall structure near the river bank. A weir is provided at the end of the discharge tunnel. The elevation of the top of this weir was determined by two factors: (1) the elevation of the water in the discharge tunnel must be such as to maintain the suction of the condenser, and (2) the discharge tunnel must be kept at least partially full of water even when the station is not operating, because

it is the source of suction for the fire pumps. The outfall structure was designed to serve as a stilling basin and utilizes concrete blocks to dissipate the energy of the water and thus avoid eroding the discharge canal.

14–2.6 Outdoor electric equipment structures. The station service and main transformers are just north of the eastern portion of the turbine-generator building. The transformers are founded on the same compacted select fill that is used for the turbine-generator building, since there must be no differential settlement between the transformer foundations and the building. North of the transformers are two transmission towers. The select fill extends far enough to support the compression legs of the towers. The tension (or north) legs of the towers exert uplift forces on the foundations. Therefore it was not considered necessary to replace the clay under the foundations with select fill.

The upper soil strata in the switchyard area south of the New Cumberland and Pittsburgh Railway, shown in Fig. 14–1, are composed of fine sands and some rounded gravel. Because the switchyard structures are light in weight, it was possible to limit the soil-bearing pressure to 1 ton/ft^2, thus keeping settling within acceptable limits.

The switchyard structures, built up mainly of light steel angles and rolled sections, are galvanized to protect against corrosion. All structural connections are made with standard bolts.

14–2.7 Miscellaneous facilities. Other facilities required as part of the station include the access road and parking facilities, railroad sidings, sewage disposal, security fencing, employees' entrance and gatehouse, and yard grading and drainage.

The road leading from the highway generally follows the eastern property line. As it approaches the railroad, it turns west along the railroad and enters the plant area proper over a grade crossing. The road circles the nuclear waste disposal area and terminates at the screenhouse (Fig. 14–1). A bridge over the railroad was considered, but a grade crossing was chosen because of the lower cost and the light railroad traffic. There are parking facilities for 125 cars near the plant entrance.

The plant is served by one railroad siding with two branches. One branch, through a series of switchbacks which lower its elevation about 11 ft, runs onto the turbine-generator building deck, where it is accessible to the gantry crane. The other branch enters the fuel-handling building at elevation 734.5 to serve the nuclear plant.

The sanitary sewage disposal system has two leaching wells and a drainage field in the sand and gravel stratum. This system was originally designed with sufficient capacity to carry the load during construction. Now that operation has begun, its capacity is more than adequate.

Atomic Energy Commission security regulations require that nuclear installations be protected with security fencing and adequate lighting. It was impractical and unnecessary to fence the entire property. In accordance with AEC regulations, an 8-ft fence surrounds the plant area proper at a minimum distance of 50 ft from any building.

The gatehouse at the employee entrance also serves as headquarters for the security guards. Anyone leaving the plant must pass through a radiation detector installed in the gatehouse. The detector minimizes the possibility of spreading contamination which may be present on the clothing, possessions, or persons of those leaving the plant.

The yard around the plant and eastward has been graded and seeded to create a finished appearance and to prevent erosion. The finished elevation of the yard is generally 734.5, except the area north and east of the turbine-generator building, which is at elevation 706. The storm water and roof drainage is intercepted by a drainage system and discharged to the river.

14–3. Electrical Design Features

14–3.1 Generating equipment. The main generator, a 4-pole, 3-phase, 1800-rpm machine designed for outdoor installation, is rated at 100,000 kw, 85% power factor, 15.5 kv; it is conventionally cooled with hydrogen at a maximum pressure of 30 psi. The generator stator is inherently weatherproof, since it must be pressuretight to contain the hydrogen; however, special features incorporated in the design make the entire generator structure weatherproof.

Hydrogen, rather than air, is used for internal cooling because it results in:

(1) Reduced windage and ventilating losses because of the low density of the hydrogen gas.

(2) Increased output per unit volume of winding because of the high thermal conductivity and heat-transfer coefficient of hydrogen.

(3) Increased life of the insulation on the stator windings because of the absence of oxygen and moisture in the presence of corona.

(4) Reduced windage noise because of the low density of hydrogen.

The guarantee capability of the generator is 94,125 kva at 0.5 psig hydrogen pressure and 117,650 kva at 30 psig hydrogen pressure. Corresponding kw ratings are 80,000 and 100,000 kw. The main terminals are brought out through six gastight porcelain bushings. Rotor and stator windings are insulated with Class B materials. The generator is designed to withstand 20% overspeed without mechanical injury.

Field excitation for the main generator is provided by a 325-kw direct connected, air-cooled exciter. With full load on the generator, the exciter supplies approximately 738 amp at 352 volts.

The commutator and brush rigging arrangement are designed so that brushes can be replaced during normal operation without dropping load. A walk-in housing over the exciter facilitates such routine replacements.

The speed-of-response factor of the exciter is better than 0.5 sec. The field and armature windings are Class A insulated and are designed for a maximum temperature rise of 40°C at rated load conditions and ambient temperature.

The exciter was designed so that it could be paralleled with a spare exciter. All excitation load can be transferred to a spare exciter (if ever one were installed) smoothly and without dropping load. However, the transfer should be made when the generator voltage is under manual control, i.e., with the voltage regulator cut out.

The generator is designed to operate at rated kva, frequency, and power factor from 5% below to 5% above the rated voltage of 15,500 volts. Automatic voltage control is obtained by a Westinghouse "Mag-A-Stat" voltage regulator, which is the latest static type designed for use with high speed response excitation. This regulator controls the voltage of the main generator by varying the excitation in the main exciter through the medium of the exciter control fields. It has a rated sensitivity of $\pm 0.5\%$ of normal voltage, and is capable of initiating corrective action without "hunting" immediately after the line voltage has departed from normal. The static voltage regulator equipment operates in conjunction with the main exciter, which is equipped with a self-exciter main shunt field, a battery excited stabilizing field, and two separate exciter control fields. The regulator is frequency compensated to maintain voltage within $\pm 5\%$ of normal voltage over a range of 75 to 150% of normal (60 cps) frequency.

14-3.2 Station service power. The station service system distributes power from the main generator and/or from the Duquesne Light Company system to all the plant electrical auxiliaries. It may also be connected to smaller capacity emergency sources when power is not available from the main generator or the system. The basic station service system, shown in Fig. 14-4, consists of two 10,000-kw transformers and four 2400-volt bus sections of metal clad switchgear. The emergency or standby sources consist of (1) a 450-kw diesel generator unit which can be connected to either of two 2400-volt buses, and (2) a 1000-kva transformer fed from the Duquesne Light 23-kv system, which can be connected to another of the 2400-volt bus sections. Interlocking permits only the diesel generator or the emergency transformer to be connected to the 2400-volt buses at any one time.

As might be expected in a plant of this type, the station service system differs considerably from that of a conventional coal-fired power station. The problems involved can be divided into two categories:

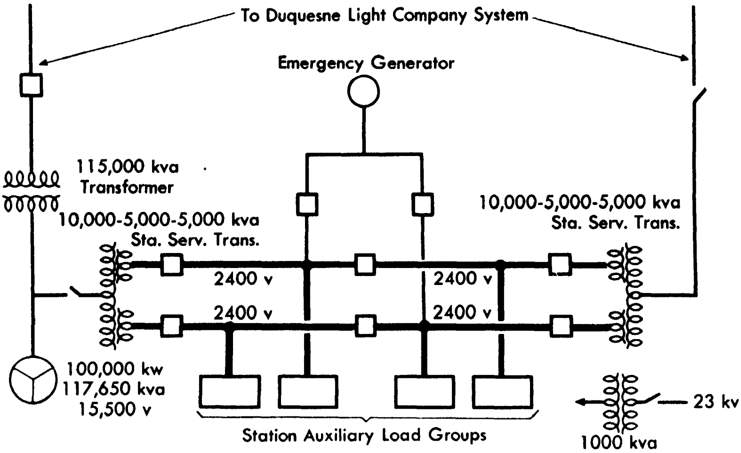

FIG. 14–4. Station service system; main single line diagram.

(1) Those encountered in serving a load of approximately 10,000 kva in which the highest utilization voltage is 2400 volts, and at the same time providing acceptable voltage drop on load transfers.

(2) Those encountered in providing adequate reliability, including power for emergency shutdown, and meeting other requirements imposed by the reactor plant.

These two requirements will be discussed in order. The largest loads in the reactor plant are the four main coolant pumps, which require about 1250 kw each when operating at full speed. Because of the temperatures involved, they are insulated with Class H material; it is considered inadvisable to extend the operating voltage beyond 2400 volts. The loads in the turbine-generator portion are larger than would be expected for a conventional 100,000-kw unit. The total anticipated demand for the full load is approximately 9800 kva, with about 2100 kva being served at 480 volts. Many arrangements involving such things as dual voltages, duplex reactors, 4160/2400 volt auto transformers, and parallel transformers were studied before the scheme shown in Fig. 14–5 was developed. The main features of this design are the three-winding transformers with special impedance so that they perform the same as two half-size, half-impedance transformers paralleled on the high side only. The transformers are rated at 10,000 kva with 9% impedance from the high voltage to each low voltage winding and with 18% impedance from low voltage No. 1 to low voltage No. 2 winding; all values are on the 10,000-kva basis. From the standpoints of fault currents and voltage drops, the transformer acts the same as two 5000 kva units each with $4\frac{1}{2}$% impedance and paralleled on the high side only. Voltage calculations indicate a drop of about 10% for a 50% load transfer and about 18% for a 100% transfer. The cost of

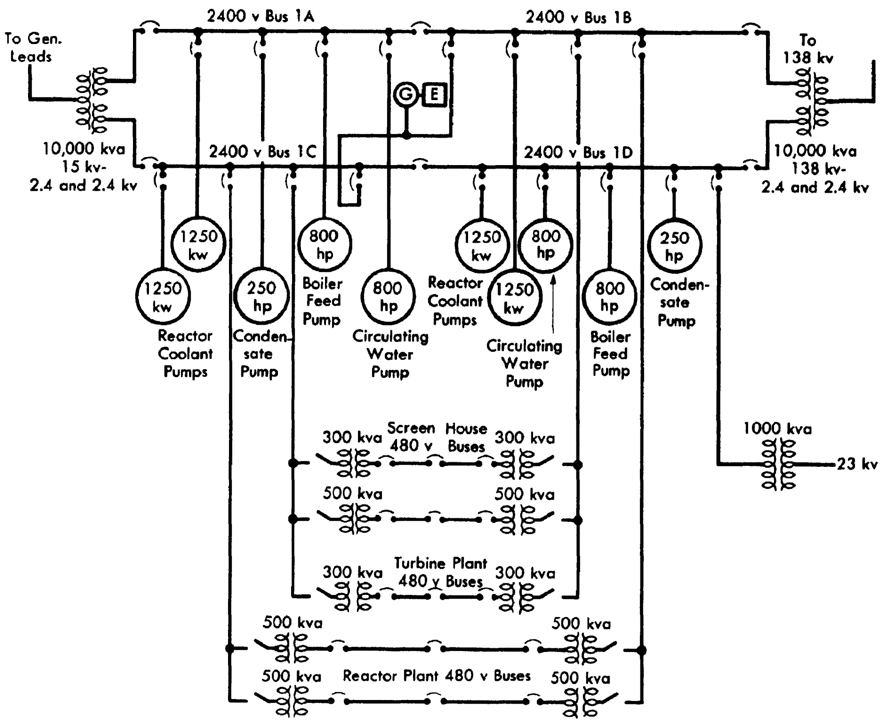

FIG. 14-5. Auxiliary power; single line diagram.

these transformers is about 10 to 12% more than a two-winding transformer of the same rating, and is more economical than two half-size banks.

The question of service reliability was given much consideration. A study of the station service requirements for the reactor plant indicated the need for high reliability and flexibility in the power supply system. To accomplish this objective, four 2400-volt bus sections were provided, with one main coolant pump per section. One other requirement for extremely reliable station service power is that for AC control power. Many of the control devices in the reactor plant require alternating current. These devices are normally serviced by direct current from the station batteries in a conventional plant.

The 2400-volt buses will normally be operated with the tie breakers open, so that half the load is on the system source and half on the generator source. Automatic transfer equipment is provided on all four sections. The ultimate objective is to be able to transfer all station service load without causing reactor scramming.

The No. 1 station transformer is a 3-phase, 60-cycle, oil insulated, self-cooled 10,000-kva transformer with a 15,000-volt delta connected high

voltage winding and two 2400-volt delta connected low voltage windings. The 15,000-volt winding is tapped to the main generator leads. The two 2400-volt windings are connected through a 2000-amp segregated phase metal clad bus duct to the 1A and 1C 2400-volt buses in the turbine room basement.

The 138-kv station transformer is similar to the No. 1 station transformer, except that the high voltage side is designed with a 138,000-volt grounded wye winding. The 138,000-volt winding is connected to the main switchyard by overhead conductors. The two 2400-volt windings are connected with a 2000-amp segregated phase metal clad bus duct to the 1B and 1D 2400-volt buses inside the turbine room basement.

There are four 2400-volt buses, or bus sections (see Fig. 14–5) which consist of Westinghouse 5-kv metal clad switchgear units. All main and tie automatic circuit breakers are Westinghouse Type 50-DH-250, with 2000-amp, 150,000-kva interrupting capacity at 2500 volts. All automatic circuit breakers in the feeder circuits are Westinghouse Type 50-DH-250, with 1200-amp, 150-kva interrupting capacity at 2500 volts.

The 480-volt turbine buses and transformers are in the turbine room basement. They consist of General Electric 600-volt, metal clad switchgear units assembled with a 2400/480 volt transformer as part of each bus unit. The 1A and 1B buses are each supplied by a 500-kva, 3-phase, delta-delta connected, 60-cycle, 2400/480 volt ventilated dry transformer with a high voltage load break disconnect switch. The 1C and 1D buses are similar, but their transformers are 300 kva.

The 480-volt screenhouse buses are fed by two outdoor transformers on the screenhouse roof. The associated switchgear and the 1A and 1B buses are inside the screenhouse. The transformers are General Electric 300-kva, 3-phase, delta-delta connected, 60-cycle, 2400/480 volt sealed dry units each with high voltage load break disconnect switches. The buses are General Electric 600-volt metal clad switchgear units similar to the 1C and 1D turbine buses.

The 480-volt reactor buses are fed by four outdoor transformers on the roof of the auxiliary power room. The four reactor buses are inside the auxiliary power room. The transformers are Westinghouse 500-kva, 3-phase delta-delta connected, 60-cycle, 2400/480 volt sealed dry units, each with a high voltage load break disconnect switch. The buses are Westinghouse 600-volt metal clad switchgear units.

14–3.3 Main transformer. The main transformer steps up the generator voltage output from 15,000 to 138,000 volts. It is rated at 115,000 kva, 3-phase, 60-cycle, and is a forced oil, air-cooled type. The high voltage winding is a 3-phase wye, with solid grounded neutral. The insulation is 115 kv, 550-kv basic insulation level, graded to 15 kv neutral. The low

voltage winding is a 15,000-volt, 3-phase, delta winding insulated 15-kv, 110-kv basic insulation level. Lightning arresters are provided on the high voltage side.

The transformer is designed to operate solidly connected to a turbine-generator unit rated at 15,500 volts. There are no provisions for disconnecting the transformer from the generator during the period when the machine is being warmed up and brought up to speed. The generator is brought up to speed with the excitation, which is required to produce 100% no-load voltage at full speed. The excitation remains on the machine from practically zero speed to 110% speed and then back to normal speed for synchronizing. During shutdown, the excitation for 100% no-load voltage remains on the machine until the speed is quite low. This transformer was designed to withstand these conditions without either exceeding rated temperature rise or vibrating excessively. The solid connection between the generator and the transformer is a common practice among utilities; it eliminates a circuit breaker between the generator and the transformer.

14-3.4 Switchyard. The switchyard distributes and controls the generated power into the Duquesne Light Company system. When the generator is shut down, it accepts power from the system.

The switchyard consists of a main 138-kv bus and four feeders equipped with switching devices. One of the feeders delivers power to the main bus from the generator, another extends from the main bus to the 138-kv station service transformer, and the remaining two feeders leave the switchyard as transmission lines. The transmission line feeders and the feed to the main transformer are each switched by Westinghouse Type GM-5A oil circuit breakers rated at 138 kv, 1200 amp continuous and 5,000,000 kva interrupting capacity.

For safety, an interlock scheme uses keys and locks to ensure opening and closing of equipment in proper sequence. Some of the interlock features are:

(1) The transmission line breakers must be open before keys are released for opening the line side and bus side disconnect switches.

(2) The air circuit breaker on the low voltage side of the 138-kv station transformer must be open before the 138-kv station transformer air break switch can be opened.

(3) The main transformer oil circuit breaker must be open before a key is released to permit operation of the 138-kv main transformer disconnects.

When switchyard equipment is closed, the reverse order must be followed.

Fire protection for the switchyard is provided by two fire hydrants delivering water from the treated water head tank.

14–3.5 Plant direct current system. Current for the essential controls in a power plant is usually supplied by a battery to ensure service continuity during plant emergencies which may involve the loss of AC power. The normal and emergency DC sources for the plant are three sets of lead acid cell batteries. The DC system supplies direct current to the reactor and turbine plants for controlling and operating such equipment as switchyard and station service relays and circuit breakers, motor operated valves, alarm systems, emergency oil pumps, turbine and reactor controls, and instrumentation.

The direct current sources are the No. 1 control battery, the emergency control battery, and the reactor plant control battery. Each battery has its own motor-generator charger, battery charging voltage regulator, and control system. The batteries float on their respective chargers continuously to maintain full charge; they discharge only when heavy loads are imposed on the DC bus. In the event that the AC power for the motor-generator set is lost, the chargers automatically disconnect from their respective batteries by means of a reverse current trip. The DC sources are ungrounded and are not operated in parallel.

Normally, all control circuits in the turbine-generator plant are supplied from the No. 1 control battery. Grounded control circuits are placed on the emergency control battery until the faults have been cleared. The No. 1 control battery, in addition to supplying normal loads, is adequate to serve the demands of an emergency shutdown resulting from the loss of AC power. It must supply control circuits for the duration of the emergency and still be able to close the 138-kv circuit breakers to restore AC power to the station. In establishing the ampere demand requirements for sizing this battery, it was assumed that it might be necessary to close two 2400-volt air circuit breakers simultaneously during the last minute of the shutdown. This would require a momentary current of 180 amp. Normal power station practice usually provides a period of 8 to 12 hr as the length of time the control battery must supply emergency power. On the basis of these requirements and conditions, a 60-cell Exide type EMP-17 battery was selected.

The emergency control battery must, in the event of loss of AC power, be able to serve the combined emergency lighting load of both the turbine-generator plant area and the reactor plant area. The turbine-generator plant requires about 8 kw and the reactor plant takes about 16 kw of emergency lighting load. The 60-cell Exide type FME-17 battery selected for this service is able to serve the load for 2 hr 19 min. The period for which the battery can serve in an emergency can be prolonged by switching off all lights not absolutely required.

14–3.6 Control room. The reactor, turbine-generator, and electrical auxiliaries are controlled from a common centralized control room. The

entire operation, including generator control and electrical switching, is conducted from this control room. There are some small local panels elsewhere, such as the water treating panel, the circulating water screen-house panel, the hydrogen control panel, and the turbine startup panel, but even these have permissive interlocks operated from the control room.

After making a study of control panel arrangements, the Duquesne Light Company engineers settled on a compromise between the conventional design and a completely graphic design. All instruments and controls were located on the panel in a functional and logical array to represent, as closely as possible, the flow of energy from left to right.

The main controls are located on a three-section console board (see Fig. 9–2). The left section controls the reactor, the center section controls the turbine, and the right portion controls the generator and station service system. Graphical representation, utilizing colored symbols and mimic bus, is used throughout. All instruments required for operating purposes are mounted on an inclined surface at the rear of the console. On a three-section instrument panel, separated by a 36-in. aisle from the console and directly behind it, are the necessary recorders, annunciators, position indicators, ammeters, etc. The 138-kv substation control panel is to the right of the generator section. The instrument panels frame into the control room ceiling and separate the main control area from the auxiliary control area. In the auxiliary control area are grouped the controls and instrumentation which are used or observed intermittently. Many systems are prepared for operation from the auxiliary control area (for example, initial water filling and the selection of auxiliary pumps). Also in this area are the control panels for the 480-volt circuit breakers in the plant. It was decided to group most of these controls to facilitate reducing auxiliary load in case emergency power must be supplied from the diesel generator.

Originally, time-saving automatic devices were considered for tripping out unwanted loads during an emergency. Later studies indicated that if station service power is lost, ample time is available to trip off the unnecessary loads manually before switching over to emergency power. Thus it was decided to leave this function under the control of the operators.

14–3.7 Relay room. The relay room, located directly beneath the control room, contains panels for protective system equipment for the reactor section, the turbine-generator section, and the transmission lines. The equipment in this room is automatic and does not require operator supervision. In line with normal practice for central station design, the relay room was located below the control room because of the large number of cables that must run from panels in the control room to the relay room. The relay room is ventilated with filtered air to prevent dust from entering

and fouling the relay equipment. The telephone equipment and plant intercommunication system are located in the relay room.

14–3.8 Communication systems. Efficient operation of a complex power plant requires a good system of communication within the plant and also between the plant and the rest of the system, particularly the system operator's office. To achieve this objective requires a number of communication systems, which will be described separately.

Page-party line system. This system provides intraplant communication for both the reactor and turbine-generator sections by means of handset and speaker stations located throughout the plant. Each handset has a "page-party line" switch which an operator places in the "page" position to page an individual or to give general instruction over the speakers. When an individual is paged, both persons place their switches on the "party-line" position and converse without being heard over the speaker. All the preamplifiers and power amplifiers are in a three-section panel in the relay room. The equipment was designed and built by the Gai-Tronics Corporation to meet audio coverage requirements specified by the Duquesne Light Company. If AC power fails, the feed to the communication panel is automatically transferred to a DC to AC motor-generator set. The power supply for the motor-generator set is the No. 1 control battery. This emergency power supply was installed because it was considered very desirable to be able to issue instructions to people throughout the plant if loss of AC power should be accompanied by a serious plant emergency.

Direct line system. This system, which includes several handsets and a control box, provides communication between the access cubicles in the reactor plant, the turbine startup panel, and the control room. Direct line communication between the above points is necessary to avoid dialing through an exchange and also to eliminate the possibility of lack of needed communications due to busy lines. Either a single or party line conversation can take place over the direct line system.

Bell system. This system provides communication between the station and the Bell Telephone System. Bell telephones are located in several offices in the station. The calls are handled through a switchboard when the receptionist is on duty and from the control room at other times.

PAX system. This system provides communication both within the Shippingport station and throughout the Duquesne Light Company. Automatic, dial type telephones are provided, and any Public Address Exchange (PAX) system telephone can be dialed directly.

System operator system. A direct line provides communication between the system operator's office and the control room. Since the system connects only these two locations, immediate communication is always available.

Radio. A radio is installed in the control room for emergency communication between the system operator's office and the control room. Power for the radio is supplied by the station service system or the emergency power system.

14-3.9 Lighting. Holaphane reflectors furnish general lighting of operating areas. They have given full satisfaction in all recently built Duquesne Light Company power plants.

Fluorescent lamps were used throughout the offices. Special attention was given to the control room, because the visual demands upon the operator are acute. Normally, operators have sufficient time to assimilate visual information transmitted to them, even with very poor or very low intensity light. However, during system emergencies or a sudden breakdown of major plant equipment, these men must in a very short time gather much information about the condition of the plant; they must check recording instruments, annunciators, supervisory indication lights, and other instruments in the control room for this purpose. Control room lighting is designed to enable the operators to see and gather all the necessary plant operating information quickly and accurately. By literally filling the room with light, the many objects to be viewed would, in effect, receive illumination from many light sources.

Desired lighting is achieved by using an installation whereby the entire ceiling becomes a lighting fixture. Rigid sheets of corrugated acrylic plastic are suspended well below the structural ceiling, to create a plenum chamber over the entire area. Rows of fluorescent lamps, mounted in this chamber at a normal spacing of 3 ft, give an average over-all intensity of about 80 foot-candles. However, at strategic locations (near vertical panels in particular) spacing was reduced to 18 in. to provide higher lighting intensities.

Outdoor driveway and security lighting is provided by conventional 370-watt, 120-volt multiple street lights, bracket mounted 37 ft above ground level. These lamps, 24 in number, were spaced at a normal span of 125 ft around the perimeter of the plant yard, giving a minimum intensity of 0.2 foot-candle over a band 60 ft wide.

14-3.10 Station grounding. The ground system was designed to keep equipment and structures at an equal potential or at safe voltages so that at no time is there danger to personnel or to equipment. All equipment frames and metal structures are grounded to the building steel, and the building steel in turn is grounded by heavy copper conductors to a low resistance earth ground. The noncurrent-carrying parts of all the major outdoor equipment, such as the power transformer, oil circuit breakers, switchyard structures, and towers, are provided with at least two separate paths to ground.

The first step in designing the grounding was a series of ground resistance tests at the station site.* Measurements at each test station were made with two reference ground rods and a sampling rod driven to a final depth of 14 ft.

The grounding grid for the main building, the reactor, and the turbine-generator plant consists of a 500 million circular-mil (mcm) copper cable encircling the building below grade, and connected every 20 ft by a "Cadweld" connection to the foundation sheet steel pilings. The main building ground system is also connected to the screenhouse ground by two 4/0 copper cables, which run below grade and connect to the steel pilings on the condenser water intake tunnel about every 20 ft. The screenhouse ground grid consists of a 4/0 copper cable which encircles the screenhouse and is connected to the foundation steel pilings every 20 ft.

The grounding grid in the switchyard consists of a 500-mcm cable, buried 1 ft below finished grade, encircling the outdoor equipment, with a cross tie through the center of the yard. Near this cable, Copperweld ground rods, 10 ft long, are driven on 25 ft centers and connected to the grid. The 500-mcm grid is connected to the main station ground by two 500-mcm cables in the main cable run. All structures are connected to the ground grid with a minimum of two connections. The fence around the switchyard is grounded separately with a No. 3 copper wire buried 18 in. below ground, connected to every other fence post, and tied to a ground rod every 50 ft.

The security fence is grounded about every 40 ft to a 10-ft ground rod; it is also connected to the main ground system at several points. Where the fence is crossed by the 138-kv lines, a heavy ground cable runs parallel to the fence and is connected to 10-ft ground rods and a fence post every 20 ft.

14–4. Mechanical Design Features

The unique character of the Shippingport reactor and the related thermal cycle imposes certain complications in the mechanical design of the turbine-generator portion of the station. Consequently, there are a few significant deviations from recent trends associated with modern conventionally fired installations. The following review points out some of the salient considerations given to the major equipment selection and design criteria.

14–4.1 Cycle arrangement. The thermal cycle for Shippingport is designed on a unit basis; that is, one heat source serving one turbine with

* This ground testing was done with James A. Biddle Company's equipment in accordance with their instruction book, "Megger Ground Testing Instruments."

no provision for steam crossover between this unit and any future unit. Three bleed points are provided on the turbine, with the extracted steam flowing to three closed feedwater heaters. All deaeration is performed in the main condenser by means of deaerating spray pipes in the condenser neck to produce proper atomization.

Included in the cycle (in the order of steam flow, condensate flow, and boiler feed flow) are the reactor, steam generators, turbine, condenser, condensate pumps, steam jet air ejector, condensate cooler, hydrogen and oil coolers, low-pressure feedwater heater, low-pressure heater drain pump, boiler feed pumps, intermediate-pressure feedwater heater, and high-pressure feedwater heater. The external moisture separator, designed to remove moisture from the steam entering the low-pressure turbine section, is in the intermediate extraction point. Demineralizers are included to provide high-purity water for makeup and for polishing the condensate and reactor coolant water.

The thermal cycle is a regenerative saturated steam cycle based on the varying pressure characteristics which result from the use of a constant average coolant temperature of 525°F for reactor control. Steam conditions at the steam generator are 600 psia dry and saturated at full load and 880 psia dry and saturated at no load. Wet steam, extracted from the turbine at three points, heats the feedwater. The drains from the high-pressure feedwater heater cascade to the intermediate-pressure feedwater heater; those from the moisture separator are taken directly to the intermediate-pressure feedwater heater. Drains from the intermediate-pressure feedwater heater are subcooled and cascaded to the low-pressure feedwater heater, where they pass through a deaerating section of the heater and into the suction of the low-pressure heater drains pump. This pump then returns the heater drains into the condensate stream. During emergency outage of any of the heaters the drains can cascade directly to the condenser.

If dry and saturated steam at 600 psia expands normally through the turbine without moisture removal or superheating, the moisture content of the steam at the exhaust end could approach 18 to 21%, depending upon load and exhaust vacuum. The problems associated with moisture can be eliminated by reheating the steam at an intermediate stage or mechanically removing the moisture in the steam by locating a separator at an optimum intermediate stage. On the basis of numerous studies made to determine the most desirable thermal cycle for Shippingport, the mechanical method of moisture removal was selected. If the cycle arrangement of reheat is compared on the Mollier diagram with external moisture removal, the two methods appear quite similar. However, the mechanical means of moisture removal, rather than reheat, was selected because the equipment cost was less and it was the simpler system. Conventional practice is to utilize reheat at elevated steam conditions when

the added cost and complexity of reheat can be justified by improved performance. This was not the case here.

14–4.2 Turbine-generator unit. The turbine is a single-shaft, high-pressure condensing unit directly coupled to a hydrogen-cooled generator and a main and pilot exciter. Although the initial core will provide steam for 60,000 kw net electrical generation, the turbine has an ultimate capability of 100,000 kw to accommodate future improvements in core design. This capability is based on initial steam conditions of 545 psig dry and saturated and $1\frac{1}{2}$ in. Hg back pressure.

The turbine was designed to have maximum efficiency at 80,000 kw gross load. A result of this requirement, the underlying reason for which is one of future plant improvement, is an improvement in partial load performance as compared with a design for maximum efficiency at full load or 100,000 kw.

This unit, designed for outdoor service, was selected because of the saving in housing costs. Weathertight walk-in enclosures are provided at both the turbine and exciter ends to facilitate maintenance and repair.

The normal operating speed of 1800 rpm was selected to minimize blade erosion problems that are aggravated by moisture present in the steam. Adoption of the 1800 rpm speed, rather than 3600 rpm, increased the physical size of the machine somewhat; however, the improved blade life, reduced leaving losses, and larger permissible pipe areas for passage of steam to and from the moisture separator offset the size disadvantage. To limit the moisture in the blade path, one stage of moisture separation external to the turbine was provided. Extensive tests under operating conditions indicated that a centrifugal type moisture separator would be satisfactory. The moisture separator is mounted beneath the turbine-generator unit to provide a compact, simplified piping arrangement. Turbine design, in conjunction with the moisture separator, provides for adequate protection from the eroding effects of high-velocity, moisture-laden steam. In regions where 6% or more moisture is present and where blade tip velocities are more than 900 fps, the rotating blades are protected by facing the leading edges with stellite strips. Likewise, at all points where sealing between stationary and rotating carbon steel or cast iron parts is required or where steam impinges directly upon carbon steel or cast iron, the surfaces are protected by weld-depositing a suitable erosion-resisting alloy. Three stages of extraction from the turbine provide steam to the feedwater heaters, consistent with the results of the thermal cycle studies.

The steam-sealed glands selected for this unit are preferred over water-sealed glands because they avoid contamination of the condensate and at the same time permit oxygen removal during the starting sequence for the turbine unit.

The steam chest valve arrangement admits all steam to the impulse stage for loads up to 80,000 kw. Above 80,000 kw additional valves permit the required steam to bypass the impulse stage.

14-4.3 Moisture separator. The separator elements of a Centrifix moisture separator are contained in a cylindrical shell. Entering moisture laden, steam first contacts the vanes of a cone-shaped or helicoid tuyere unit. This tuyere imparts a swirl to the steam, forcing the heavier moisture particles outward to the wall of the separator shell while the lighter vapor stays nearer the center of the cone. The heavier moisture particles are swept forward into an annular race, where all the solid particles are trapped and ejected through the drain. The exhaust pipe, which extends into the separator shell, has vanes on the internal end to prevent fluid from flowing upstream and destroying the vortex pattern set up by the tuyere. The moisture removed from the steam is collected in and discharged from the integral drain chamber built into the bottom of the exhaust end of the separator.

The selection of this type of moisture separator for the wet steam cycle required confirming data; it was therefore decided to test a prototype separator under simulated service conditions. The test was performed at the Duquesne Light Company's Stanwix Station in November 1954, utilizing a noncondensing turbine. Results established the relationship of pressure drop, separator size, and efficiency level throughout the operating range of the turbine.

14-4.4 Condenser. The single-pass steam condenser is welded to the turbine exhaust flange. It is a horizontally mounted unit of 70,000 ft^2 surface area, with a deaerating hot well and divided water box capable of condensing 817,000,000 Btu/hr with 114,000 gpm of circulating water at 85°F. The hot well has a capacity at normal water level of 9400 gal; it is equipped with a high and low level alarm and level controller. The hot well is sized to provide a minimum of 5 min protection to the condensate pumps. The 30-ft long condenser tubes are made of Admiralty metal; the tube sheets are Muntz metal. These alloys were chosen in accordance with conventional practice. The 30-ft long tubes were selected because they are the longest tubes available that would satisfy the design conditions for the condenser.

This condenser is much larger than a similar unit in a conventional plant of the same kw rating. The reason for the difference in size is that the Shippingport unit must handle about 50% more steam than conventionally designed condensers. Consequently, larger condensate and circulating water pumps are needed.

The condenser deaerates the feedwater during normal operation. Since

steam-sealed shaft glands have been provided on the turbine, full vacuum can be established with the turbine on turning gear. During this period the condensate pump will recirculate condensate to the condenser and complete deaeration will be accomplished as the condensate re-enters the condenser through flash nozzles.

14–4.5 Feedwater heaters. The three feedwater heaters in the system are of the horizontal, U-tube type. They are conventional except for the low-pressure heater, which was specially designed to eliminate oxygen and reduce the possibility of stress corrosion cracking.

Dissolved oxygen and chlorides are significant factors in accelerating the stress corrosion cracking of stainless steels. Since any equipment that may operate below atmospheric pressure, such as the low-pressure heater, is a potential source of air in-leakage, all water-side surfaces were examined critically. For design purposes, the relationship of tube cross section to shell cross section was stipulated to provide at least 20% more area than that which had been calculated. This liberal margin provides area for steam to assist in deaerating drips and drains. In addition, a deaerating section with trays further assures the return of oxygen-free condensate to the system.

Consistent with the design criterion that maximum efficiency occur at 80,000 kw load, the low- and high-pressure heaters were designed for a 5°F terminal difference. The intermediate-pressure feedwater heater was designed for a 7°F terminal difference, while a 15°F difference was established for the integral drain cooler.

Since this was to be an outdoor plant, it was desirable to minimize the number of openings through the operating floor; therefore horizontal rather than vertical feedwater heaters were selected. Horizontal feedwater heaters offered a better arrangement in the plant because the tube bundles or shell could be removed, serviced, and inspected without the necessity for providing hatches in the turbine deck.

14–4.6 Boiler feed. Two motor-driven boiler feedwater pumps satisfy the basic design criterion of 60,000 kw net electrical generation. Normally, it is Duquesne Light Company practice to install three boiler feed pumps sized so that with any two pumps operating, the maximum rating of the turbine and steam generators will be realized. The third pump serves as a spare. In the Shippingport plant, a single pump can provide enough water to the steam generators to produce the specified 60,000 kw net electrical generation; the second pump is used as a spare. The installation of a third feed pump was deferred until such time as the reactor was able to produce the anticipated 100,000 kw.

These pumps also provide water to the safety injection system for the reactor coolant. This is made possible by arresting the flow to the

steam generators and diverting the water to the reactor coolant system piping.

The boiler feedwater pumps are vital components of the station, and it is extremely important that they be reliable and constantly available. For increased reliability, one pump is operated from the main generator while the other is operated from the 138-kv system. An automatic recirculation system protects the pump at low flows.

Feedwater control must be fast in operation to maintain the water level within normal limits during load changes, so as to nullify the effect of swell. Thus, consistent with conventional practice, three-element feedwater control was selected for this plant.

Normally, these three metered elements, steam flow, water flow, and water level, are combined pneumatically to operate the regulating valve. This accomplishes the following:

(1) Proper proportioning of metered water flow to metered steam flow, regardless of variations in water pressure or of valve port design.

(2) Readjustment of the water flow to maintain the boiler drum level within normal limits regardless of load.

The Shippingport installation is complicated by the fact that the steam pressure varies inversely with load (conventional stations operate at constant pressure). This pressure variation introduces an error in the water level measurement due to the temperature change in density of the water within the boiler drum. Consequently pressure compensation was specified for this application.

14-4.7 Water treatment facilities. Water, the most important single item upon which the entire station is dependent, is needed as reactor coolant, for boiler feedwater, for the fuel-handling canal, for cooling purposes, for sanitary and drinking uses, etc., each having its own easily identifiable specification. It should be noted that the reactor coolant system requires large quantities of high-purity water, regardless of whether the turbine is operating. Thus the water treatment facilities at Shippingport must of necessity have greater capacity than those at conventional stations.

During the construction phase of the project, the water requirements were satisfied by two wells drilled adjacent to the river. Some consideration was given to the use of these wells for the permanent water supply to the plant. However, since wells tend to become plugged and since the area water is undesirably hard, it was decided to use the river as the source of supply and to install a water treating plant.

In describing the water treatment system, the following nomenclature is used to define the state of the water:

Treated water—water that has passed through the clarifier, and has been chlorinated and filtered.

Softened water—treated water that has passed through a salt regenerated cation exchanger.

Demineralized water—softened water that has passed through an anion and cation exchanger (this is similar to distilled water).

The treated water plant contains a high-rate, solid-contact clarifier with a mixing flume, a flocculation zone and a sedimentation zone, automatic pressure filters, and a clear well with two treated water pumps. The clarifier has equipment for lime and alum or Ferrisul feeding. Design of the clarifier has been based upon the worst river pollution conditions and has been specified to produce water with a maximum turbidity of 5 ppm.

The effluent from the clarifier flows into an open clear well. The water in the clear well is chlorinated beyond the break point and is then pumped through the pressure filters. Each filter has a capacity of 200 gpm, one-half the capacity of the clarifier. The filtered water, still under pressure, flows to a 60,000-gal treated water head tank. The treated water can be chlorinated after it has been filtered, but this "post chlorination" is a safety measure only; normally, it will not be performed.

The treated and chlorinated water enters the head tank at the top and leaves at the bottom. It is retained at least 4 hr, long enough for the chlorine to kill all bacteria. The water leaving the 60,000-gal tank is suitable for human consumption.

The softened water is produced by passing the treated water through two styrene base, automatically regenerated softeners. This water is used for demineralizer regeneration and hot water requirements such as showers, laundries, etc. The softeners can produce 320 gpm of water with a maximum hardness, as $CaCO_3$, of 2.0 ppm.

The final step in the water treatment process subjects the water to mixed-bed demineralization; this produces the high standard of purity specified for makeup for the reactor coolant grade water and boiler feedwater. The demineralizer effluent, based upon soft water influent with a maximum hardness (as $CaCO_3$) of 2.0 ppm, will have a maximum conductivity of 1.5 micromhos/cm, maximum chlorides of 0.3 ppm, and a pH between 8.3 and 6.5 at 25°C.

The use of evaporators to provide this grade of water was considered. However, because steam was unavailable and would have been costly to obtain in the amount required, it was decided to use demineralizers. There was no question whatever that an evaporator could produce water of the specified purity, except for the problem of chloride removal.

The demineralized water system contains two completely automatic mixed-bed demineralizers, a 3-psig deaerator, acid and caustic storage tanks, and a neutralizing tank.

The makeup demineralizer normally provides makeup water of condensate quality. The effluent from this demineralizer passes through a heat

exchanger and the deaerator, then to the Duquesne Light Company 50,000-gal water condensate storage tank.

The condensate demineralizer is normally used for polishing service on the condensate system. Approximately 150 gpm of condensate is bled and pumped through the demineralizer; part of the flow is returned to the condensate system, while the remainder is directed to the 50,000-gal primary water storage tank.

Piping and valving are so arranged that the condensate demineralizer can be used for makeup duty whenever the normal makeup demineralizer is out of service.

Supplementary Reading

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CHAPTER 15

SITE DESCRIPTION AND DEVELOPMENT

CHAPTER 15

SITE DESCRIPTION AND DEVELOPMENT*

15–1. General Location

The Shippingport Atomic Power Station site is on the south bank of the Ohio River in Shippingport Borough, Beaver County, in southwestern Pennsylvania. It is approximately 25 miles northwest of Pittsburgh and about $4\frac{1}{2}$ miles east of the Ohio-Pennsylvania border.

15–2. Property Description

The outline of the property and the location of the plant on the site are shown in Fig. 15–1. The plot of land, about 420 acres, is irregular in shape, having a mean length of about 5500 ft and a mean width of about 3500 ft. The property is divided by the New Cumberland and Pittsburgh Railway Company right-of-way, a 100-ft wide strip of land that runs generally east and west. The shaded area in Fig. 15–1 represents the land that is controlled as to use and occupancy. In order to protect the general public in the unlikely event of an incident at the plant, these properties will not be used for any purpose that permits access or assembly of the general public.

The hydrology, meteorology, seismology, population distribution, and other characteristics of the site pertinent to evaluation of reactor hazards have been compiled and studied and are discussed in WAPD-SC-547.†

Figure 15–2 is a topographic map of the area, showing the generally hilly nature of the land. The ground surface slopes up from the river, the normal pool elevation of which is 662.6, in a south and southwest direction. Near the west boundary line, a steep valley wall rises rapidly to an extensive plain at elevation 1165. Near the south property line the ground rises to a ridge at elevation 1125 and then slopes down to the valley of Peggs Run, a small stream that enters the property in the southwest corner at about elevation 850. From this corner, Peggs Run flows to the east property line, which it follows to the Ohio River.

There are two relatively flat but gently sloping areas, or terraces, on the property. The lower area (*A* in Fig. 15–2) is in the northeast corner,

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* By R. J. McAllister, Duquesne Light Company.

† "Description of the Shippingport Atomic Power Station Site and Surrounding Area," WAPD-SC-547. (Available from the Office of Technical Services, Department of Commerce, Washington 25, D. C.)

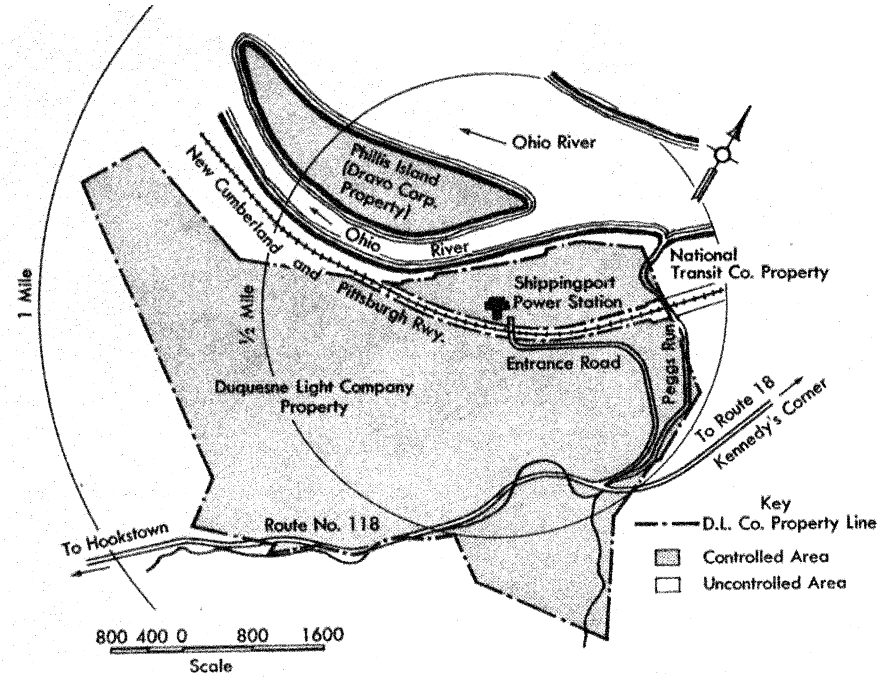

FIG. 15–1. Station property plan.

near the river. This area is about 200 ft in width and varies in elevation from about 685 to 690. It is bounded on the north by a more steeply sloping area, which extends to the river. Its southern boundary is a steep slope, approximately 50 ft high, that runs generally parallel to the river and merges with the valley wall at the west property line. South of this latter slope is another flat area (*B* in Fig. 15–2). This area is roughly triangular in shape, being about 1200 ft wide at the east end and extending about 2600 ft westward. It merges with the steep valley wall on the west and the south. The ground surface of this area slopes gently upward to the south and west, varying in elevation from about 730 to about 760.

The soil profiles in the vicinity of the building area are shown in Figs. 15–3 and 15–4. These were drawn from data obtained from soil borings. Throughout this area, the bedrock is a thinly bedded hard gray shale with occasional sandy shale and sandstone members. The bedrock slopes generally downward to the north and east. Deposits of well graded sand and gravel of the same type and character cover the entire area above bedrock to varying depths. On the upper terrace this sand and gravel layer is about 80 to 100 ft thick. It is overlain by 10 to 20 ft of generally fine sand with some silt. On the lower terrace about 30 to 40 ft of sand and gravel cover the bedrock. This sand and gravel undoubtedly covered the

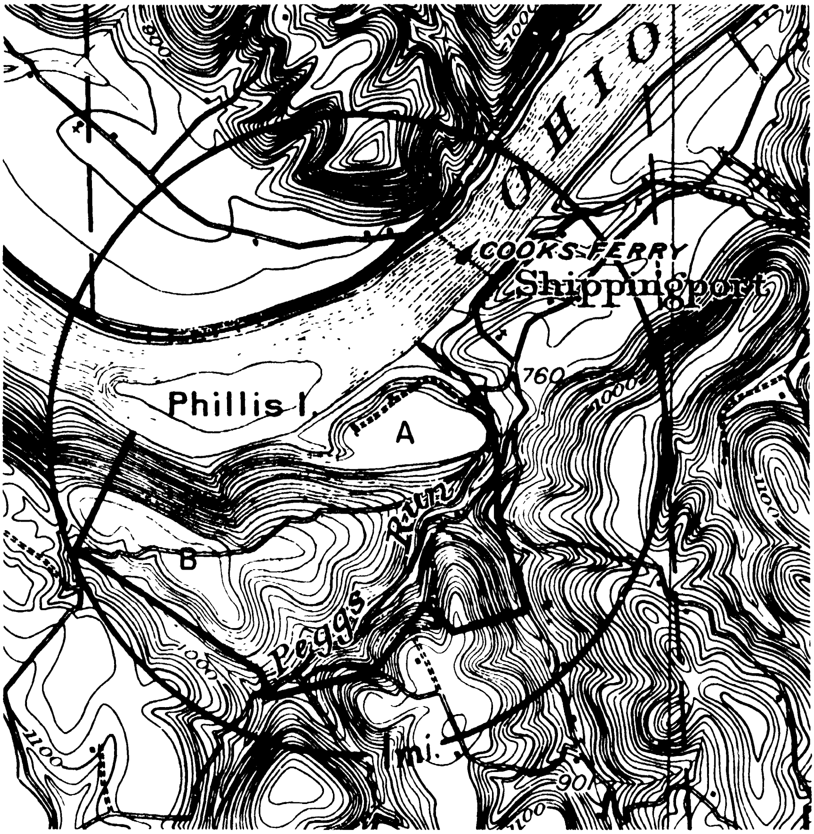

FIG. 15-2. Topography of the PWR site.

area to greater thickness at one time, but apparently it was eroded and subsequently replaced with 20 to 30 ft of fine sand, clay, and silt. The slope between the lower terrace and the river is composed of silts and organic materials over fine sands, clays, and silts.

The site is located on a back channel of the Ohio River. This back channel, which is about 400 ft wide, is separated from the main channel by a low-lying uninhabited island known as Phillis Island. The river flows in a southwest direction as it approaches the site, then turns and flows away from the site in a northwest direction. There is no established harbor line at the site. As stated previously, the normal pool elevation of the river is 662.6. This elevation is maintained by Dam No. 7, a wicket type dam one-half mile downstream from the site. According to flow duration curves, this pool elevation can be expected about 70% of the time. Elevations above the normal pool level can be expected about 15% of the time and elevations below it about 15% of the time. The low water

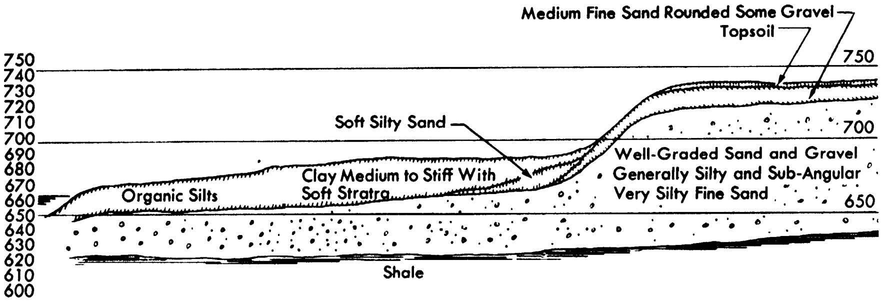

FIG. 15–3. Soil profile in northwest-southeast direction east of building area.

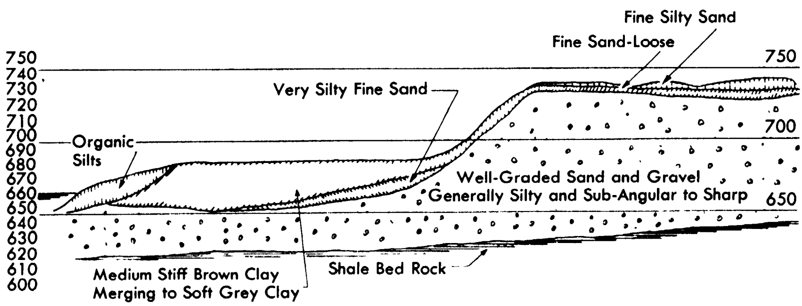

FIG. 15–4. Soil profile in northwest-southeast direction through building area.

elevations occur, as a rule, not during periods of low flow, but after periods of very high flow. At such times, the level of the river may be allowed to recede to about $3\frac{1}{2}$ ft below normal pool to facilitate closing the wickets. A new dam is now under construction, about $16\frac{1}{2}$ miles below the site, which will replace three of the old style dams, including Dam No. 7, and will raise the pool level at Shippingport to about 664.5. It is expected that this dam, being of the gated type, will be better able to maintain the normal pool.

The Ohio River is subject to periodic flooding. This was considered in the design of the plant and is discussed in Chapter 14.

The lowest flow ever recorded at Dam No. 7 was 2790 ft^3/sec. The flow duration curves indicate that approximately 23,000 ft^3/sec are available 50% of the time and at least 50,000 ft^3/sec 25% of the time.

15–3. LOCATION OF PLANT ON SITE

In general, the location of the plant on the site (Figs. 14–1 and 14–2) was chosen to make the maximum present and future use of the land. Economics precluded the use of the lower terrace. This area is subject to frequent flooding, which would have increased the difficulties of construc-

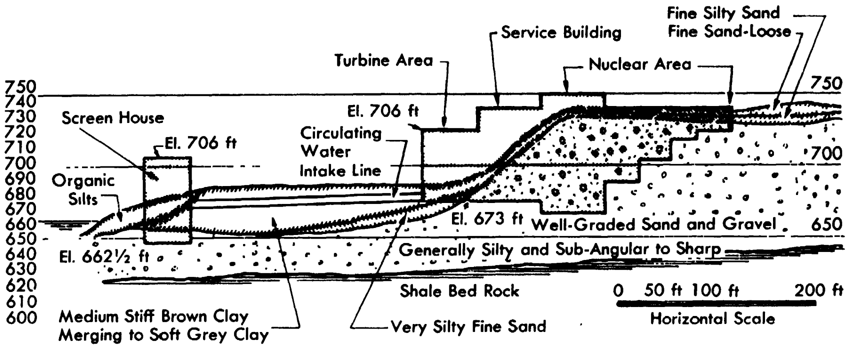

FIG. 15–5. Approximate location of foundations in northwest-southeast soil profile.

tion and would have added considerably to the cost of protecting the Station against flood water. It would have been necessary to found the structures at a low elevation to obtain good bearing. Lowering the foundations, with the design flood elevation fixed at 706, would have greatly affected the design of any walls or portions of the building subject to hydrostatic pressure, and would have greatly increased the cost.

As can be seen in Fig. 15–5, the Station was located partly on the upper terrace and partly on the slope to the north of that terrace. This north-south location was chosen for the following reasons: (1) Good foundation conditions were available for the most part at the elevations desired; (2) the buildings could be placed high enough to simplify flood protection and at the same time not so high as to disproportionately increase the cost of pumping circulating water; and (3) because of the terrain, much of those portions of the plant which require shielding could be placed underground, thus taking advantage of the earth for shielding. The east-west location was chosen as far west as possible, consistent with the narrowing of the usable portion of the property by the railroad right-of-way to the south and the low lying area to the north. This permitted an arrangement which provides for the maximum present and future use of the property.

As part of the site development, the upper terrace was graded from its original ground surface elevation of 730 to 760 to a uniform elevation of 734.5. This extensive grading operation provided a large level area on which were located construction and permanent railroad tracks, construction buildings, storage facilities, etc. At the same time it yielded large quantities of fill which were used to extend the leveled section to greater limits than those of the original terrace. The portion of the level area remaining after construction of the plant provides space for future expansion.

15-4. Arrangement of Plant Facilities

Fuctionally, the station is divided into two parts; the nuclear section, which is essentially the steam producing portion, and the turbine-generator section, the primary function of which is the generation of electrical energy using the steam produced by the nuclear section. Except for certain detached facilities, the buildings of both sections are combined into one integrated structure, the arrangement of which is shown in Fig. 16-2.

The nuclear section of the plant is built around the reactor. The fuel-handling building, which contains facilities for handling and storing radioactive materials, runs south from the reactor chamber. Boiler chambers Nos. 1 and 2 flank the reactor chamber to the west and east respectively. The auxiliary chamber is located just north of the reactor chamber with its longest dimension in the east-west direction. Various nuclear laboratories and facilities to service the nuclear plant are grouped in the reactor service building. This building is located east of the fuel-handling building and south of No. 2 boiler chamber.

The turbine-generator plant is built to the north and adjacent to the nuclear plant buildings. The major building is the turbine-generator building, which contains the electrical generation equipment and associated auxiliaries. The turbine-generator service building, which houses offices and equipment for control and coordination of operation of the entire plant, is located east of the auxiliary chamber and south of the east end of the turbine-generator building.

Because of their functions, certain facilities are detached from the main group of buildings. The radioactive waste disposal area is located west of boiler chamber No. 1. The screenhouse, where circulating water is drawn from the river, is located near the river bank, north of the main buildings. A circulating water intake tunnel connects the screenhouse pumps with the condenser in the turbine-generator building, and the discharge tunnel returns the water from the condenser to the river. The electrical transmission substation which feeds the electrical output of the Shippingport Station into the Duquesne Light Company system is located south of the main building, across the railroad. Other facilities located on the plant site include the entrance road, the railroad sidings, the parking lot gate house, and the security fence.

Supplementary Reading

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3. J. W. Simpson et al., PWR Power Plant, *Westinghouse Engr.* **15**, 178–189 (1955).

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5. S. Baron and T. L. R. Williamson, Instrumentation of a PWR Atomic Power Plant, *J. Instr. Soc. Am.* **5**(1), 46–51 (1958).

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7. R. J. McAllister, Fly Ash Concrete for Shippingport Atomic Power Station, *Proc. Am. Soc. Civil Engrs.* **83** (PO2-No. 1215), 1–14 (1957).

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11. R. F. Valentine, *Hazards to the Area Surrounding PWR Due to Atmospheric Diffusion of Radioactivity*, USAEC Report WAPD-SC-548, Westinghouse Atomic Power Division, September 1957.

12. Westinghouse Atomic Power Division, *Bettis Technical Review, Reactor and Plant Engineering*, USAEC Report WAPD-BT-5, December 1957.

CHAPTER 16

ARCHITECTURAL DESIGN OF NUCLEAR PLANT

CHAPTER 16

ARCHITECTURAL DESIGN OF NUCLEAR PLANT*

16–1. Introduction

16–1.1 General plant layout. The general layout and arrangement of the buildings of the Shippingport plant were chosen to take maximum advantage of the topography, geology, and other characteristics of the site.

The principal structural and architectural features of the plant are combined into one major structure. This structure consists of two contiguous facilities, which will be referred to hereafter as the nuclear plant and the turbine-generator plant. Each has other associated facilities.

The principal features of the nuclear plant are the reactor plant container and its enclosure, the fuel-handling canal and building, the reactor service building, and the auxiliary equipment rooms. The waste disposal area and building are adjacent to this main structure. The entire Station facilities, except for the switchyard, are shown in Fig. 16–1. Figure 16–2 is a plan view of the major structure with several portions cut away at different elevations to show the relative locations of the principal facilities. Figure 16–3 is a cutaway showing the relationship of the canal to the reactor plant container and its enclosure.

16–1.2 Principles which governed layout and arrangement. The functional requirements which were major governing factors in the general plant layout were:

(1) The reactor is to be refueled by replacing either individual subassemblies of the core through ports in the vessel head or the complete core with the head removed. In either case water is to provide a radiation shield during fuel handling. The water above the reactor vessel will also provide shielding during normal plant operation.

(2) In case of complete loss of power to the Station, coolant must flow by convection from the reactor vessel to the heat exchangers to remove decay heat.

(3) There are four reactor coolant loops. Plant design capacity is obtained with any three steam generators in service. The fourth loop is, in effect, an installed spare.

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*By R. F. Devine, Westinghouse Bettis Plant, and M. Shaw, U. S. Atomic Energy Commission.

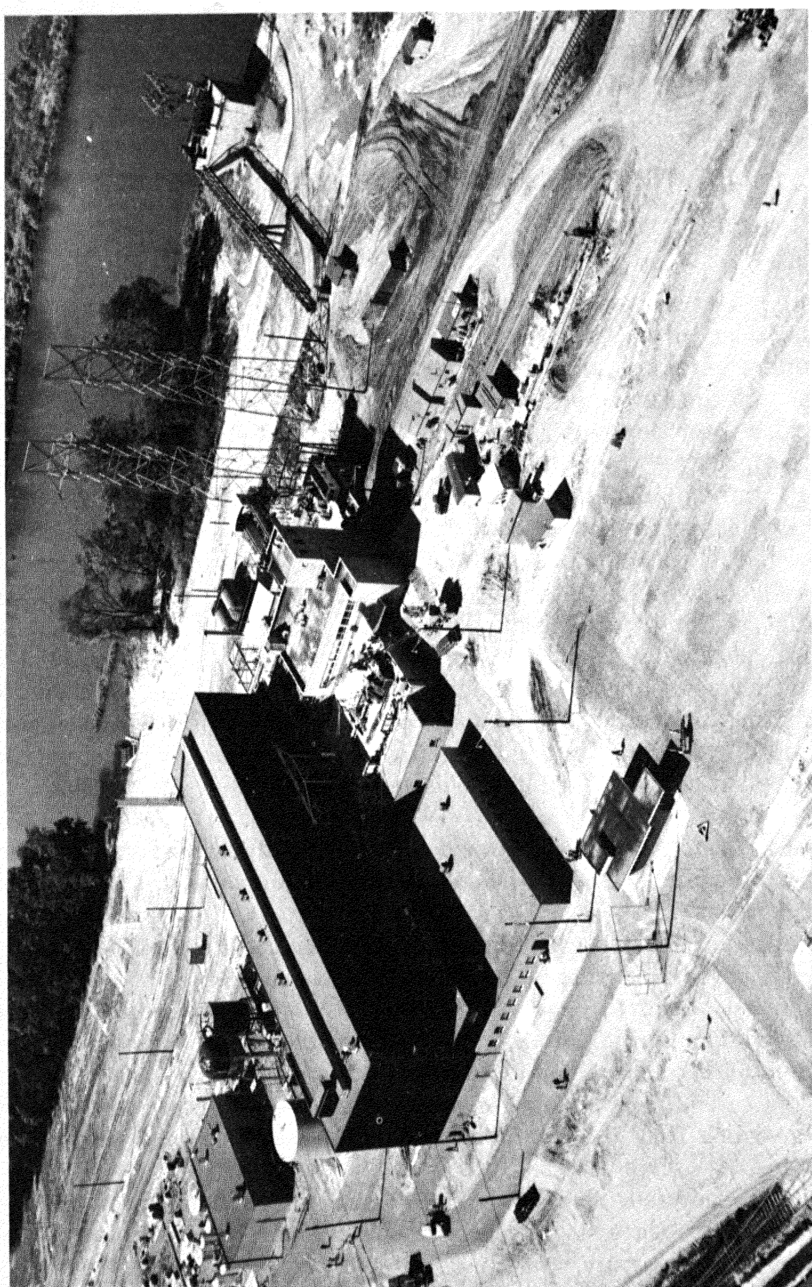

FIG. 16–1. Shippingport station.

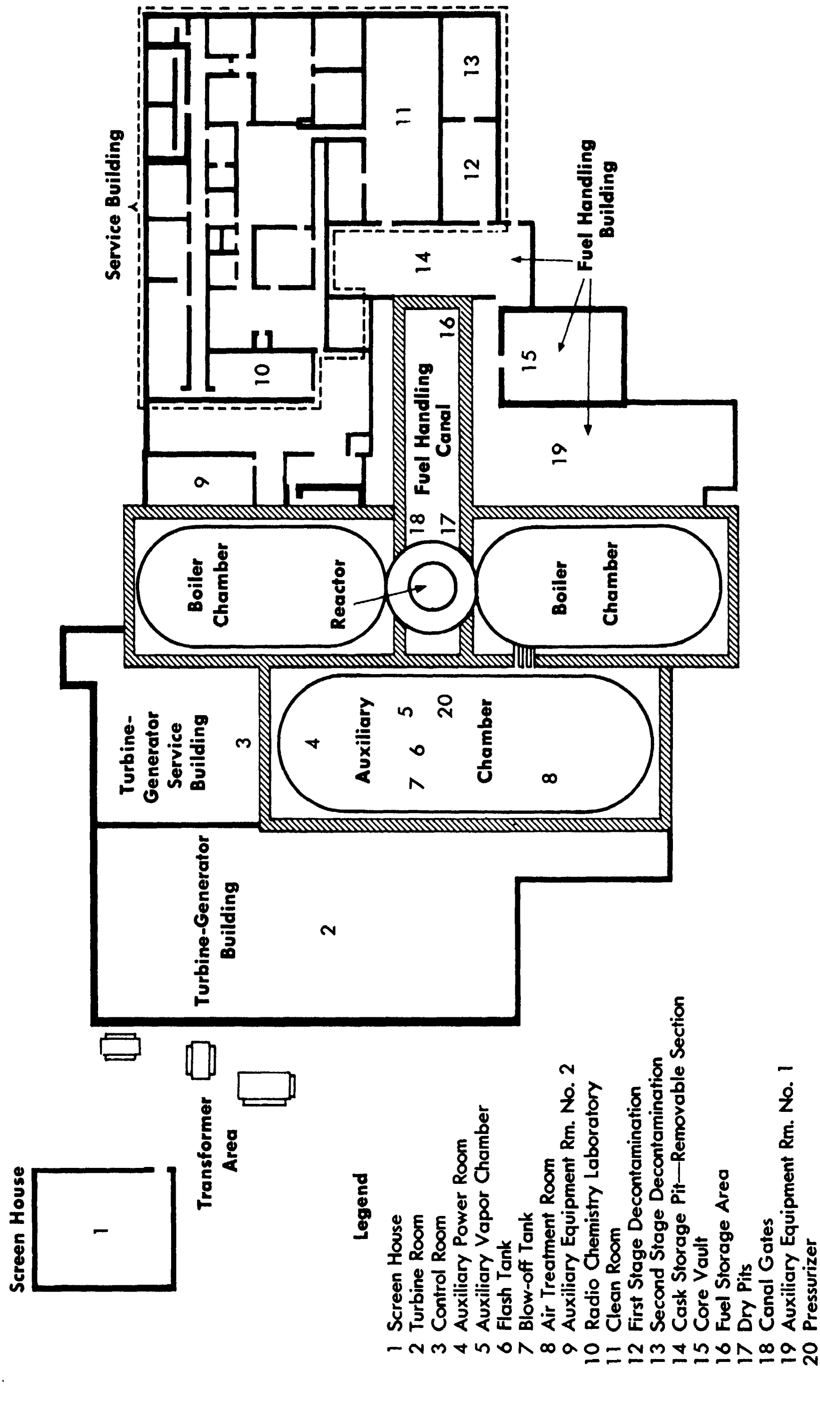

FIG. 16–2. Plan view of main PWR structure.

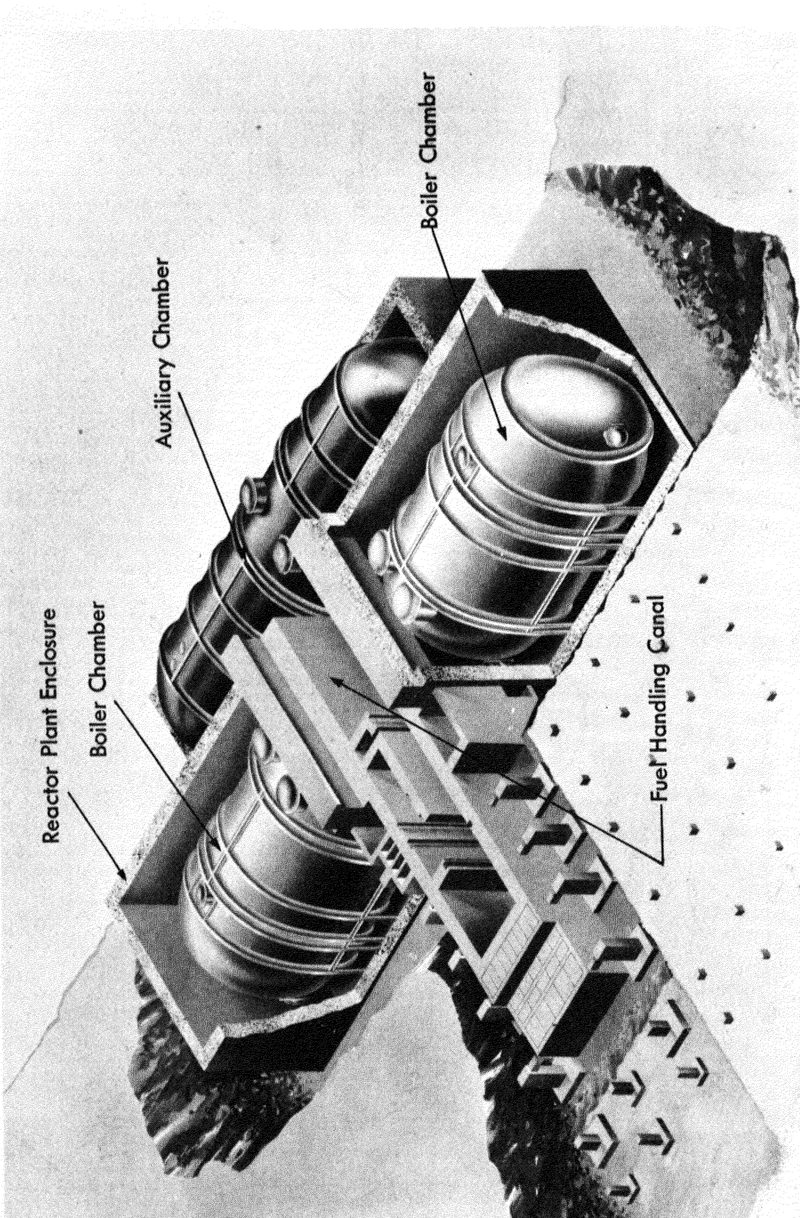

FIG. 16-3. Cutaway view of reactor plant enclosure and fuel-handling canal.

(4) The reactor plant is so housed that in case of a rupture in the reactor coolant systems — a very remote possibility — no harmful quantity of radioactive material will escape to the atmosphere.

(5) Areas adjacent to the reactor plant are shielded from radioactivity and each loop is shielded to permit certain maintenance operations while the plant is operating.

(6) Subassemblies of used cores will be removed from the core assembly under water. They will then be stored under water to allow partial decay of fission products before shipment to the reprocessing plant.

(7) Provisions are made to dispose of radioactive wastes.

(8) The Station is to be self-sufficient in laboratory services for radiochemistry and health physics.

(9) A facility is provided to decontaminate mechanical equipment.

16–1.3 Building codes and design criteria. Volume 6000, Part 6300 of the Atomic Energy Commission Manual was used as a basis for design wherever applicable. This part of the manual specifies the governing codes and criteria to be used in the planning, design, and construction of AEC facilities. The stated intent of the design criteria is "to insure that designs for construction programs will be purely utilitarian and without unnecessary refinement."

16–2. Development of the Layout and Arrangement

16–2.1 Influence of the canal on general layout. As mentioned in Chapter 4, refueling under water requires a large water filled canal extending over the reactor vessel. This canal is also needed to store and cool irradiated fuel, either a complete core or individual subassemblies. The precision manipulator type extraction crane travels the full length of the canal at floor level. A working area provided around the canal is, for convenience and accessibility, 4½ ft above yard grade. The normal canal water level is 18 in. below floor level, the water depth ranging from over 43 ft in some sections to 24½ ft in the section over the reactor. This portion of the canal is a governing feature, since this depth is needed to provide the minimum desirable shielding when irradiated fuel is handled.

Since one method of refueling is to exchange fuel subassemblies through ports in the reactor vessel head, space had to be provided within the vessel, above the core, to withdraw a subassembly from the core, move it laterally to a position beneath a fuel port in the head, and transfer it to the canal. This requirement was a factor in determining the height of the reactor vessel, and in turn its bottom elevation, since, of course, it had to be located directly beneath the canal.

16–2.2 The reactor plant container. *Selection of size, type, and arrangement.* In the early stages of preliminary plant design it was decided that all plant systems containing reactor coolant at high temperature and pressure would be contained within a pressure-tight vessel. Then, even in the very unlikely event of a rupture in any of these systems, any radioactivity that might otherwise be released to the atmosphere would instead be contained in this vessel. It also followed that this container should be surrounded by a radiation shield, both because of short-lived nitrogen-16 activity during operation and because of the radiation hazard that might exist if a system ruptured. This shielding requirement amounts to the equivalent of approximately $4\frac{1}{2}$ ft of ordinary concrete. For design purposes, the container was to have sufficient capacity to contain all vapor resulting from a complete release of all reactor coolant and the water from the secondary side of one steam generator.* Therefore, it was important to keep pipe runs as short as possible to keep the reactor coolant volume and consequently the container volume at a minimum.

This container was one of the major items of cost in constructing the plant, and along with the canal, it was a governing feature in the layout and arrangement. The container structure was placed substantially underground, since this permitted taking advantage of earth for shielding and was believed to be a safer arrangement. A further advantage of this arrangement was that the turbine-generator building would be contiguous to the main structure and at as low an elevation as possible, consistent with flood levels at the site, to keep condenser cooling water pumping costs to a minimum.

To restrict the size of the container, it was considered desirable to select the highest design pressure practicable. Since many penetrations of the container shell were required, it was decided, as a matter of judgment, that 50 psi should be about the maximum. It then fell in the category of an unfired pressure vessel and the Commonwealth of Pennsylvania regulations for such vessels became the basis for design. A welded steel structure using SA 201 Grade B firebox steel was decided upon, allowing use of steel up to $1\frac{1}{4}$ in. in thickness without stress relieving. In order to construct the container essentially underground, avoid excavation to a prohibitive depth, conserve site space along the river, and provide for thermal circulation of coolant, a multi-chambered vessel was specified.

After considerable study of various shapes and arrangements, a cylinder was selected as the most economical basic shape to meet the requirement that the container be located underground. With $1\frac{1}{4}$-in. plate and a design

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*This subject is discussed in detail in WAPD-SC-549, "PWR Plant Container Sizing Criteria" (available from the Office of Technical Services, Department of Commerce, Washington 25, D. C.).

pressure of about 50 psi, it followed that the basic cylinder should be about 50 ft in diameter. To provide space for two steam generators side by side, design considerations gave rise to two similar interconnected cylindrical chambers, each housing two of the four coolant loops. Each of these was made long enough to house most of the auxiliary equipment and shielding associated with the two loops.

To minimize pipe requirements, these two chambers were required to be as close to the reactor vessel as possible, yet to be shielded from it. Also, as pointed out before, the reactor had to be placed beneath the canal and at a lower elevation than the heat exchangers to allow for gravity circulation of coolant to remove decay heat if power failed. These factors established the elevation of the steam generator chambers.

The optimum shape of the chamber to house the reactor vessel, from the standpoint of space and economy, was a sphere. The sphere has a large removable dome on top, protruding into the canal; the dome, which must be removed to provide access for refueling, was shaped to house the control rod drive mechanisms on the reactor vessel head.

The additional container volume needed was then provided by another cylinder of the same general design as the other two, but longer, called the auxiliary chamber. It is connected with each of the other chambers by large interconnections which also serve for passage of piping, ductwork, and walkways. This chamber is largely void, but does contain the pressurizer and some other auxiliary equipment with concrete shielding surrounding them.

The auxiliary chamber was placed at a lower elevation than the boiler chambers so that the top of the concrete shielding deck over it is at the same level as the floor adjacent to the canal. This deck thus provides good accessibility between the canal area and other areas and furnishes additional work and storage area for refueling operations and equipment. On one end of the deck is the equipment for ventilation of the plant container and on the other end is the electrical switchgear room. The center portion became part of the fuel handling building. This arrangement was structurally well suited to the contour of the site and the adjacent turbine-generator building.

Container design description. (a) General. For the general arrangement of the plant container, see Fig. 16–4. The gross volume of the container is 600,000 cubic feet; the net volume, approximately 473,000 cubic feet. This container is a multi-chambered pressure vessel designed for 52.8 psig and 280°F as well as for a 3-psig vacuum. Approximately 2200 tons of steel were required in constructing the vessel.

The sphere in which the reactor vessel is located is 38 ft in diameter with a cylindrical dome (18 ft in diameter by 20 ft high) on top to accommodate the extra height of the control rod drive mechanisms.

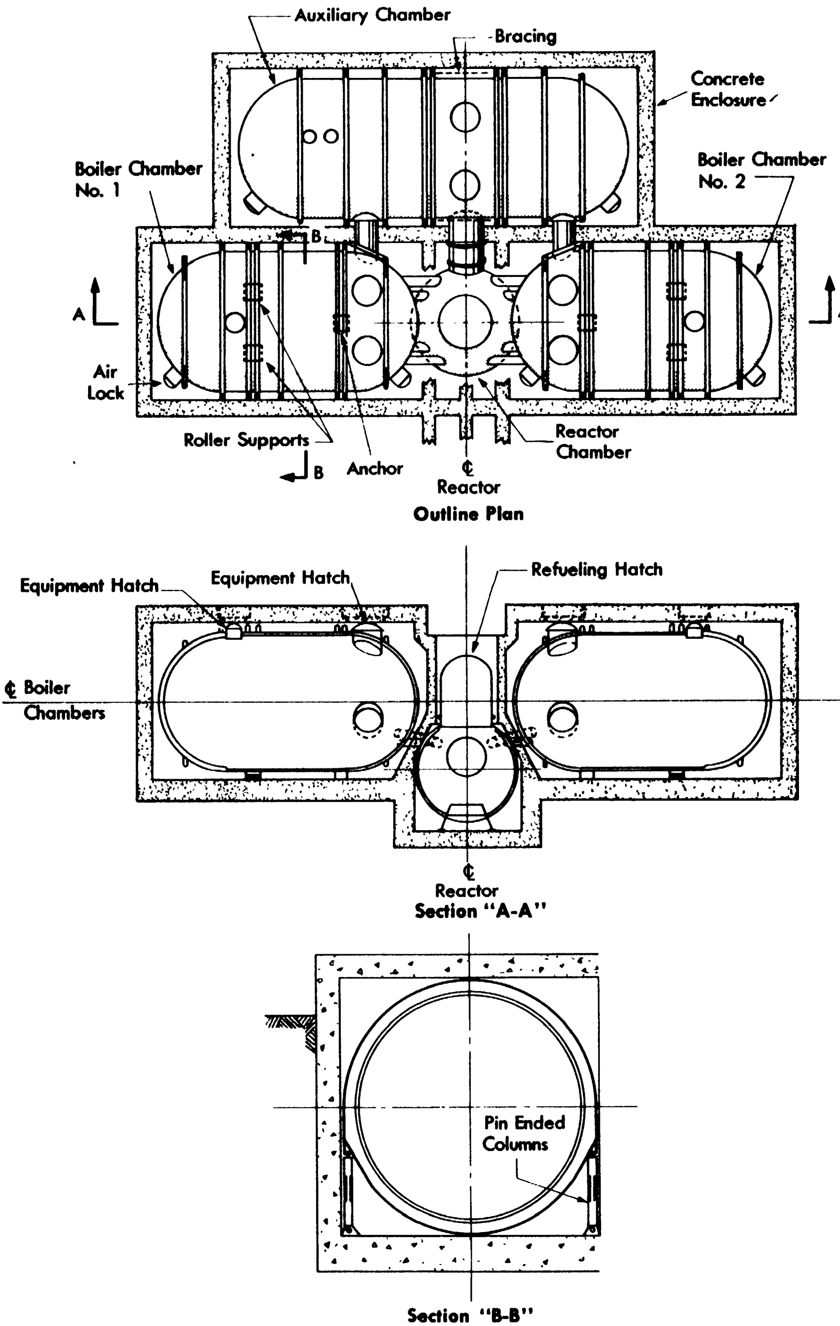


FIG. 16–4. General arrangement of reactor plant container.

Located in line on each side of the reactor chamber are the 50-ft diameter horizontal cylinders, called boiler chambers, as shown in Fig. 16-4. Each cylinder has hemispherical ends and is 97 ft long. The additional chamber, called the auxiliary chamber, is located between the boiler chambers and the turbine-generator structure. The auxiliary chamber, also a 50-ft-diameter horizontal cylinder with hemispherical ends, is 147 ft long over-all. The coolant pressurizing tank and other auxiliary equipment are contained in it. All these chambers are interconnected by several large diameter ducts.

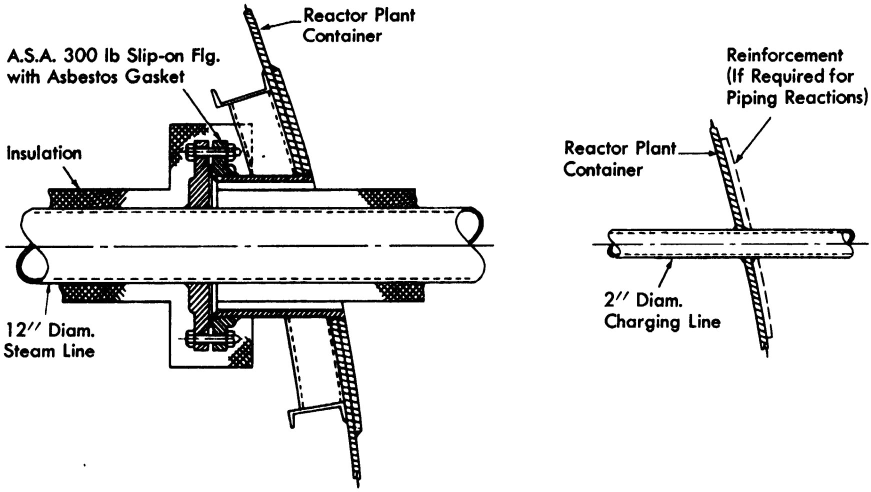
(b) Support and movement. The reactor chamber and equipment within it are supported by a conical skirt. The combined load of the reactor chamber and its contents and the water-filled fuel-handling canal above the reactor chamber is transmitted through the skirt to the concrete foundation, which forms the central anchor for the piping and the plant container system.

Each of the two boiler chambers is supported by (1) a single anchor located near the reactor end of the chamber; (2) two roller assemblies some 32 ft distant from the anchor in a longitudinal direction; and (3) a series of circumferential structural rings, each supported by a pair of columns connected to the rings.

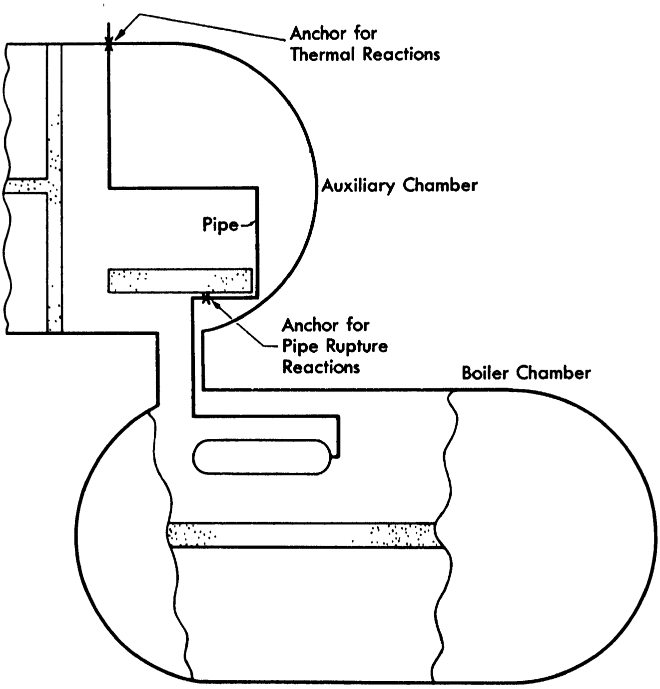
The columns are pin connected at the top and bottom, so that radial expansion of the compartment is accommodated by movement at the pin joints, whereas longitudinal movement from the anchor point is taken by flexure of the columns.

The auxiliary chamber is not anchored, and is supported entirely by similar rings and columns. The auxiliary chamber is free standing in a transverse direction by virtue of the pin connected columns, but is held in vertical position by the three large interconnecting tubes to the reactor and boiler chambers. Longitudinal motion of the auxiliary chamber is restricted by cross bracing between the two central pairs of columns. Radial or axial temperature movement is taken by the columns in the same manner as for the boiler containers.

(c) Interconnections between chambers. The three large interconnecting tubes joining the reactor and boiler chambers with the auxiliary chamber are designed primarily for the passage of expanding vapor, to allow all four chambers to become filled without excessive pressure buildup in case of a rupture in the reactor coolant system. They also carry piping and permit personnel passage between the auxiliary chamber and each of the other three chambers. The tube from the reactor chamber is 12 ft in diameter, while those from the boiler chambers are each 8 ft in diameter (Fig. 16-4). The axes of the tubes are oriented in a radial direction relative to the respective chambers to minimize stresses at the junctions with the chambers. The three interconnections are rigid in axial direction but



Typical Pipe Penetrations



Diagrammatic Arrangement of Pipe Anchors

FIG. 16–5. Plant container piping penetrations and pipe anchors.

allow translatory or shear movement of the chambers with thermal expansion of the container shell.

Eight sleeves, 3 ft 5 in. ID and fabricated of $1\frac{1}{2}$-in. plate, also shown in Fig. 16-4, interconnect the reactor chamber with the boiler chambers and allow passage of the reactor coolant inlet and outlet pipes. Each sleeve contains two expansion joints similar to those in the large interconnections. These sleeves are designed to take translational movement and axial expansion and compression.

(d) Personnel access openings. A total of six air locks are provided for personnel entry, two each for the boiler and auxiliary chambers. For greatest utility, they are located at opposite ends of each chamber. Locations of these air locks are shown in Fig. 16-4. Each consists of a cylindrical chamber 7 ft. in diameter and 8 ft 9 in. long, welded into the reactor container shell in a heavily reinforced opening. Each lock will be closed by two quick-closing pressure-tight doors, each of which can be opened or closed from either side. The doors will be so interlocked that one door must be fully closed and locked before the other can be opened. The operating mechanism of each door will also actuate a vent valve to equalize pressures prior to opening of the door. Three of the air locks normally used for access during plant operation have electrically operated locking mechanisms.

(e) Equipment access openings. The top of the container has nine openings for equipment handling. There are two 10-ft-diameter and one 6-ft-diameter equipment hatches in each boiler chamber and two 10-ft-diameter hatches in the auxiliary chamber. These hatches consist of short, flanged nozzles welded into reinforced openings in the containers and sealed with flanged, spherically dished covers which are gasketed and bolted tight. There is also the 18-ft-diameter reactor chamber dome.

(f) Container shell penetrations. In addition to access openings for personnel and equipment, many penetrations of the container shell are necessary for piping, ventilation, and electrical circuits. Consistent with pressure vessel design, the shell plate was reinforced at such penetrations.

Piping which penetrates the shell is either anchored to internal concrete shielding as near the shell as possible, or is reinforced sufficiently to take reaction forces that could occur if a pipe ruptured. All piping inside the container is designed to withstand stress associated with a minimum temperature of 280°F. All piping is externally valved in some manner to prevent any escape from the container if a pipe is broken internally. Typical piping penetrations are shown in Fig. 16-5.

All electrical penetrations are similar in design. They are basically nozzles capped with spun heads into which are welded pipe couplings for threading in conduit from either side. On the outside, a short section of conduit is welded into a flanged cylindrical junction box. Bolted to the

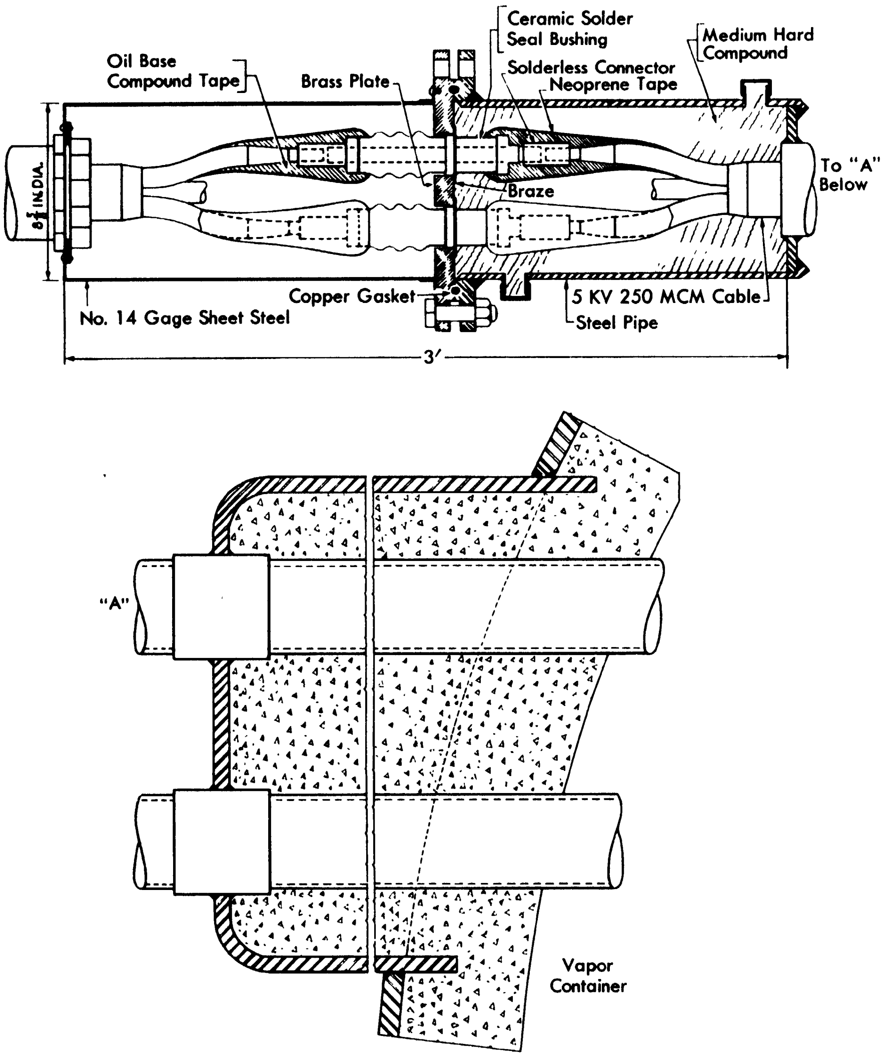


FIG. 16–6. Plant container electrical penetration.

flange, with a copper gasket, is a brass plate acting as a seal. These plates contain insulating bushings with studs for connecting cables on each side. A typical electrical penetration and stop joint is shown in Fig. 16–6.

There are two ventilation openings, each 4 ft in diameter, which are similar in design to the typical pipe penetration. External to the container are expansion joints on each to provide for container movement. One opening is for supply and the other for exhaust, and each is provided with two 48 in. hydraulically operated, quick closing, butterfly type valves.

Testing. All welded joints were radiographed during construction. Immediately upon completion of construction, the container was pneumatically tested in accordance with ASME code requirements. The pneumatic test was made at 70 psi after all joints had been hammer-tested and subjected to magnetic particle inspection. Following this testing, a leak rate test was conducted at design pressure (52.8 psi). To facilitate the construction schedule, the pneumatic and air leak rate tests were conducted in three parts. Each boiler chamber was tested separately; the auxiliary chamber and the spherical reactor chamber were tested as a unit. Then the temporary heads which had been welded in place at the interconnections between the boiler chambers and the other chambers were removed. During these leak rate tests, considerable difficulty was experienced from temperature variations of the air in the container—the container was exposed to weather conditions with air temperatures varying about 30°F between day and night, and the container top surfaces were exposed to the sun during the day. Container pressures cycled with ambient temperatures and the thermocouples installed in the container did not give true container air temperatures. Leak rates were estimated by comparing pressure readings at various times of the day when air temperatures were comparable. These air leak rate tests indicated that the combined leakage rate was about 0.15% of total contents of the container in 24 hr.

After all equipment had been installed and erected within the container and just prior to plant operation, another leak rate test was conducted with the container pressurized to 10 psi gage with air containing 0.2 vol. % of Freon-12. All potential leaks were checked with an electronic leak detector. The only leaks found were at mechanical joints and these were remedied. The principal sources of leaks were the copper gaskets at electrical penetrations. After these gasketed joints were tightened or adjusted, the leak rate was markedly reduced. Further observations then indicated the leak rate to be within acceptable limits.

The basic method used in conducting this test was to compare the absolute pressure within the container with that of an isolated reference volume consisting of 13 lengths of 1-in. copper tubing, each approximately 20 ft long, suspended at various points within the container. These were interconnected by 1/4-in. copper tubing which was brought out to the exterior of the container and connected to a manometer. The lengths of 1-in. copper tubing were placed in such a manner that the pressure within the system was judged to be a function of the average absolute temperature of the air within the container. Before the test this reference volume was carefully leak tested. At the beginning of the test the reference volume was vented to the container so that it would be at the same pressure as the container. One side of a differential manometer was then connected to the reference volume; the other side, to the container. In this manner

the leakage rate was a direct function of the change in differential pressure read at the manometer. No corrections were needed for temperature, since the reference system temperature closely approached the temperature of the air within the container. As a secondary check on the test method, the absolute pressure within the container was constantly observed, as was the temperature within the container at approximately ten different points. These temperatures were "read out" in the control room, the sensing elements being thermocouples installed to observe various temperatures within the container during plant operation. There was good agreement between the two sets of data.

16–2.3 Reactor vessel support. The reactor vessel is supported by a series of 24 lugs welded to the vessel, equally spaced around its periphery in a plane about three feet below the outlet nozzles. Each lug bears on a case hardened steel pin (4 in. in diameter by 6 in. long) positioned radially with respect to the vessel. These pins in turn bear on a steel ring ($3\frac{3}{4}$ in. thick by 6 in. wide) surrounding the vessel. The pin holes, half of each hole being in the lugs and half in the support ring, were drilled by special machines after the vessel was oriented, plumbed, and shimmed to the correct elevation. The pins were then inserted in the holes and welded to the ring at the outer ends to prevent them from sliding as the vessel expands and contracts (see Fig. 16–7). This method of supporting the vessel on a ring of radially positioned pins was chosen because it permits the vessel to expand and contract with temperature changes but to remain fixed in position and serve as an anchor for the 18-in. piping loops.

The pin support ring was welded to an inverted U-shaped ring girder, with stiffened horizontal web, surrounding the vessel. To the inner of the two vertical flanges of the girder is welded a 1-in. plate formed into a cylinder which carries the load downward to the conical skirt section resting on the heavy concrete foundation slab. The cylindrical plate forms the inner wall of the neutron shield tank. A similar plate welded to the outer ring girder flange forms the outer wall of the tank. A vertical section through the vessel, support, and shield tank is shown in Fig. 16–7.

The vessel support was designed for a direct load of 1,000,000 lb and a simultaneous horizontal force of 1,000,000 lb at the plane of the outlet nozzles.

The direct load transmitted to the support through the pins is derived from the vessel and its contents and the hydrostatic head imposed upon the vessel as the compartment above the vessel is flooded during refueling Additional load is transmitted to the conical support below the vessel from the neutron shield tank, the circular walkway around the vessel, the reactor container, and, when in place, the dome and the canal water above it.

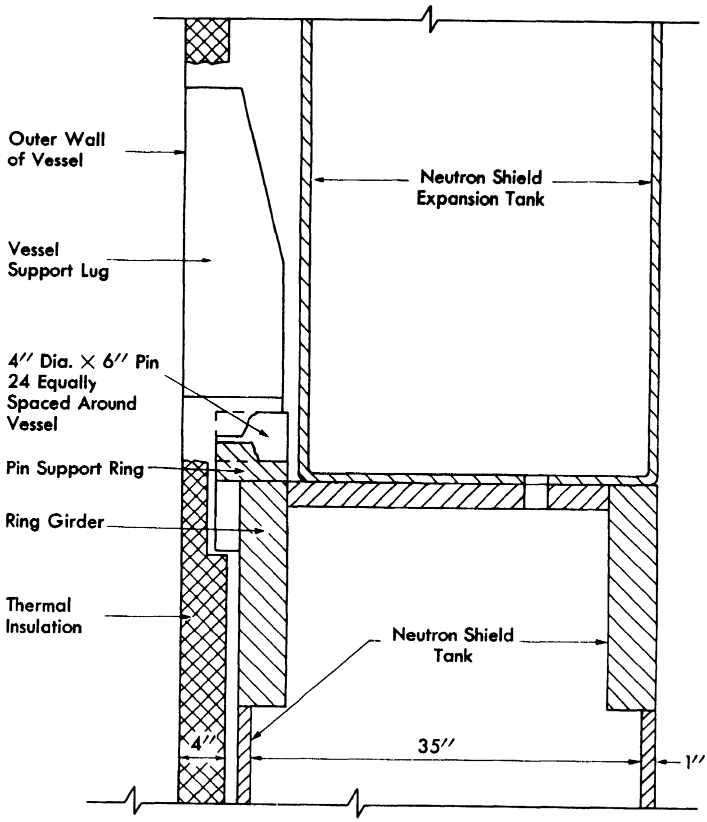


FIG. 16-7. Detail of reactor vessel support.

The assumption of 1,000,000 lb for the horizontal force acting in the plane of the upper nozzles was considered ample allowance for the effects that would be caused by the abrupt shearing of the 18-in. pipe near an outlet nozzle.

16-2.4 Fuel-handling building. A plan of the floor area of the fuel-handling building is shown in Fig. 16-8. A longitudinal section through the building is shown in Fig. 16-9. These figures show the relationship of the container and its enclosure to the canal, the fuel-handling building, the clean room, the decontamination room, and the core vault.

The fuel-handling building is 60 ft high, 51 ft wide, and 260 ft long. This building houses the canal, the clean room, the decontamination room, the heating and ventilation equipment for the building, and a storage area over the auxiliary chamber enclosure. It contains a 125-ton traveling bridge crane which is used for handling fuel and items of heavy equipment.

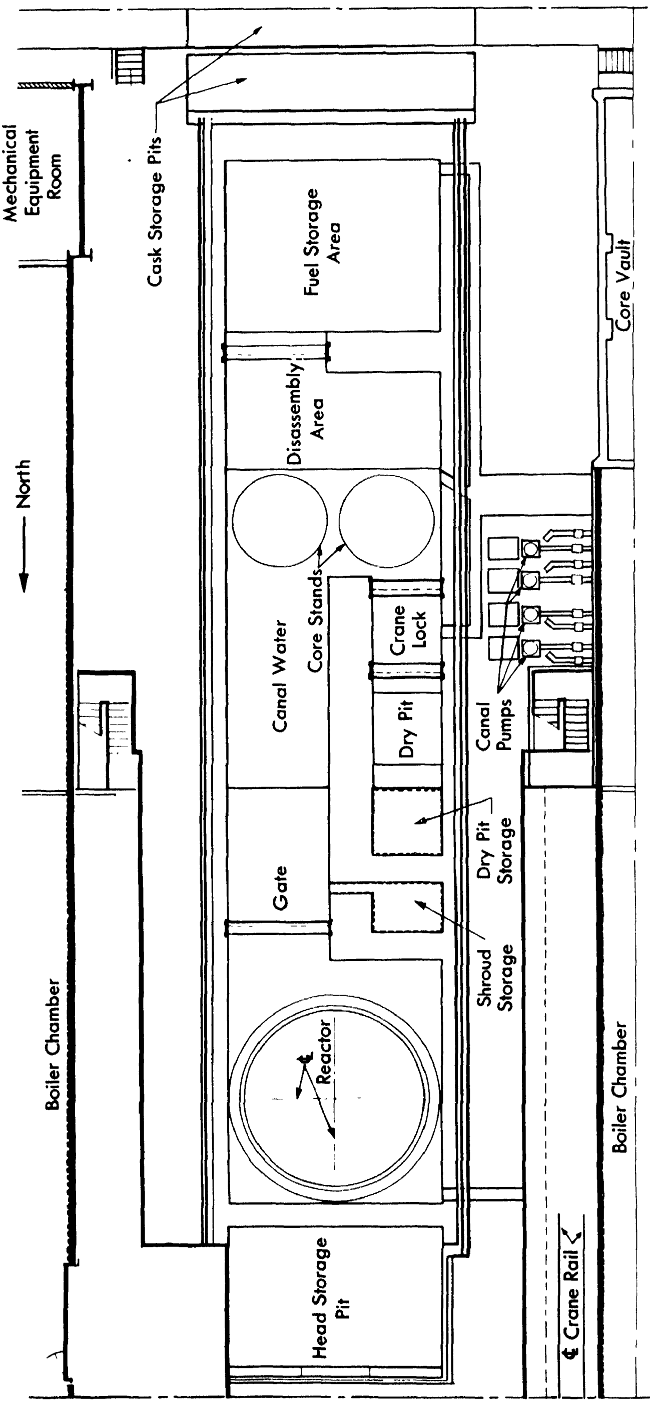


FIG. 16-8. Plan of fuel-handling building and canal.

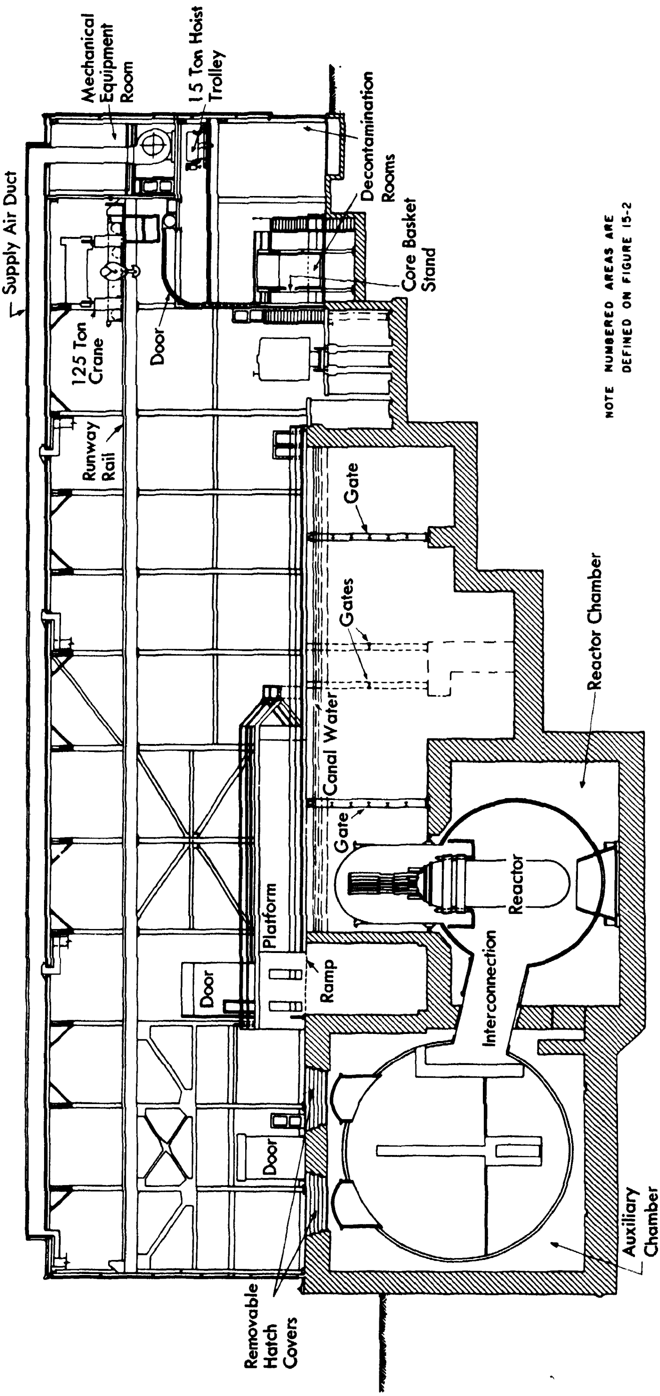


FIG. 16-9. Elevation of reactor plant and fuel-handling facilities.

This crane serves the storage area over the auxiliary chamber as well as the clean and decontamination rooms. Rail or truck shipments can be brought into the building under this crane at the south end of the building. Trucks can also enter the building from the deck over the western end of the auxiliary chamber, which is provided with a ramp to grade. The height of the fuel-handling building was dictated by the size of the bridge crane and the precision extraction crane used for refueling through the reactor vessel head. The latter has a tall mast, the length of which is dictated by the depth to which it must reach for fuel subassemblies in the reactor vessel. It is necessary that the bridge crane be high enough to clear this mast.

The *clean room* is designed for reassembly and check-out, in a dust-free atmosphere, of new cores and equipment such as control rod drive mechanisms. A slight positive air pressure is maintained in both this room and the fuel-handling area. Adjacent to the canal is a core vault for storing new fuel subassemblies.

The *decontamination room* is provided for cleaning radioactively contaminated equipment. The reactor vessel head, large pumps, or large valves can be placed in this room by the fuel-handling crane. A slight negative pressure is maintained in the room to prevent spread of airborne radioactivity.

The *canal* is 109 ft long and 22 ft wide, with a capacity of 419,000 gal, and is divided into four sections, separated by hydraulic gates. These gates are set in place and removed by the main fuel-handling crane. Any individual section can be pumped dry. One of these sections is a lock which permits the precision crane to be moved to a dry pit for maintenance of the fuel assembly extraction tool. There is also a pit adjacent to the reactor section of the canal for storing the reactor vessel head.

16–2.5 Auxiliary plant equipment housings. Contiguous with the fuel-handling building are several smaller structures for housing equipment not located in the reactor plant container.

On top the auxiliary chamber enclosure and on the west side of the building is the air treatment room, which houses the equipment for the container air cooling system. On this level to the east is the auxiliary power room, which houses the electrical switchgear for the reactor plant. Immediately next to this room, in what is known as the turbine-generator service building, is located the main control room for the combined plant.

Immediately south of each boiler chamber enclosure and adjacent to the fuel-handling building at ground level is another structure housing plant equipment. One houses pumps for the component cooling water system and heat exchangers for this and the canal water system. The other houses various equipment associated with the coolant, charging,

valve operating, and canal water systems, such as pumps, compressors, vessels, and valving.

16-2.6 Reactor plant service building. Laboratory and service areas are in the reactor plant service building, which is adjacent to the fuel handling building and readily accessible to it. In this area are located the lockers and showers for personnel, the laundry, and all laboratory facilities and related offices.

The planning phase of the design of this building was difficult — it was necessary to determine exactly what the building would have to provide. After this had been decided, the design involved, for the most part, straightforward engineering problems. Therefore, rather than a description, the functional requirements of the building will be given here. The functions of each of the principal facilities will be described separately.

Health physics facility. The health physics facility is provided to control health hazards incident to the operation of the reactor plant, particularly those from radiation and toxicity. To accomplish this function, this facility consists of the following:

(1) *Office space for health physics personnel* and for records. In addition to making radiation surveys and maintaining records thereof, the surveyors collect waste effluent samples, prepare smears of contaminated floors or other contaminated surfaces, collect air filter samples, and in general, maintain continuous surveillance over the reactor operations and repair work to ensure that health hazards are anticipated and adequate safeguards provided. The routine minor maintenance and upkeep of radiation survey instruments normally includes only battery and tube replacements, assembly of film badges and charging and reading of dosimeters, and reading and recording of developed film badges.

(2) A *radiation safety storeroom* for storing respirators, air masks, oxygen masks, protective and lint-free clothing reserves, "radsafe" signs and markers, and similar equipment.

(3) A *health physics laboratory* for preparing aliquot samples to determine the extent of radiation and toxicological health hazards present in the air, in sewage effluent, on equipment and material, and elsewhere, incident to the operation of the reactor plant. Continuing surveillance is maintained as an operational control, and the function cannot be performed elsewhere without giving it a status of secondary importance. Samples from smears, sludges, dusts, filter pads, and solutions are prepared for determining activity. Samples of urine may be examined to determine the amounts of uranium, plutonium, and mixed fission products present.

(4) A *darkroom* for rapid processing of film badges when an employee has been exposed to near tolerance doses of radiation and a decision must be made regarding further exposure. Similar quick processing is required

in cases of possible overexposure which may be indicated by an off-scale reading of a dosimeter. Processing by remote facilities would not serve the purpose satisfactorily.

(5) A *health physics counting room* for determining radioactivity of long and short lived radioactive samples (many of which are of low intensity, e.g., two to three counts per hour). The room is shielded with two-foot thick concrete walls, ceiling, and floor containing electrically bonded copper mesh screening. Labyrinth shielding is provided at the entrance.

(6) A *calibration room* for calibrating all portable radiation counters, film badges, and pocket dosimeters, using a known source. This room, located next to the radiation work space, has wood floor, walls, and ceiling to minimize back scattering of source radiations.

Operational chemistry facility. The operational chemistry facility is provided primarily to evaluate and control the reactor coolant and secondary water systems. The facility is also used to analyze radioactive waste and reactor coolant system crud samples.

Since water chemistry can change markedly as a result of age and transportation, analyses must be made as near the sampling point as practicable. Similarly, some of the anticipated radioisotopes are extremely short lived, and their presence would not be detected if much time elapsed between the sampling and analytical operations.

Accordingly, the following laboratory facilities for analysis are provided in the reactor plant service building:

(1) A *reactor coolant sampling room* for obtaining water and suspended solids samples from the reactor coolant system, necessary for operational controls. Hydrogen is also injected into the reactor coolant from this room. Shielding is required between the reactor coolant system and the operator who takes the samples in suitable shielded pressure vessels (bombs) and conveys them to the sample preparation room.

(2) A *sample preparation room*, located between the point of sampling and the radiochemistry laboratory. Here samples are prepared for analysis. This room also has a truck entrance so that shielded samples from other locations, such as sampling points in the radioactive waste disposal system, can be delivered to be prepared for analysis.

(3) A *radiochemistry laboratory* to identify and measure fission and corrosion products in the primary reactor coolant and in the ion exchangers and other system components; to detect and measure fuel element failure by water analyses for uranium, plutonium, and gaseous fission products; and to make isotopic analyses of uranium and plutonium.

(4) A *cold chemistry laboratory* for such work as is normally associated with any conventional power plant. Some of the services of this laboratory are essential for controlling the reactor plant. For this reason and because the employees operating this laboratory are the same ones who operate

the sample preparation room and the radiochemistry laboratory, it was considered desirable to locate this facility in the reactor plant service building.

(5) A *chemical calculation room* to provide suitable space for technicians to perform calculations and maintain necessary records.

(6) A *counting room* for chemistry work. A separate room is necessary since the samples handled in chemistry and health physics work are of different ranges of activity. A 1-foot thick concrete shielding wall 6 ft high separates this room from the health physics facility to prevent interference between the different sample activities to be processed.

Locker and shower rooms. The locker and shower rooms, known collectively as the change room, permit personnel who might work in an area which could be radioactively contaminated to change clothing completely and take a shower before leaving the plant, in accordance with safety regulations. The change room is divided into two areas, one considered to be "contaminated" and the other "clean." Shower facilities are located between them. The "contaminated" area has locker space for work clothing and a wash room. There is access to this area from the fuel-handling building, the radiochemistry laboratory, and the sampling room. The "clean" area has lockers for workers' street clothing and contains the necessary sanitary plumbing fixtures. There is also access to this area from the clean room. Monitors surround the exit, which is from the "clean" side of the change room, so that any radioactivity on personnel leaving the room may be detected.

Basement tank room. There are three systems of drains within the service building. One system carries nonradioactive wastes, such as those from urinals and closets, and goes to the sanitary waste system provided by Duquesne Light Company as a part of the site. Another set, known as "monitored" drains, is provided for wastes which can be, but normally are not, radioactive, such as those from lavatories, showers, and some laboratory drains. The third set, known as "special monitored" drains, would normally be expected to carry radioactivity.

The latter two systems of drains go to separate collecting tanks in a basement of the building where the waste can be sampled and monitored before being pumped to waste disposal. These two sets of drains are carried in trenches in the floor of the building for ready access. Chapter 10 discusses the processing of these wastes.

Miscellaneous. The other facilities which are a part of the service building are as follows:

(1) *Assembly room* — provides space for briefing 15 to 20 men on work programs in the reactor facility, particularly during shutdown repair operations. It may also be used as a lunch room for approximately 20 to 30 people, or for engineering conferences or training classes.

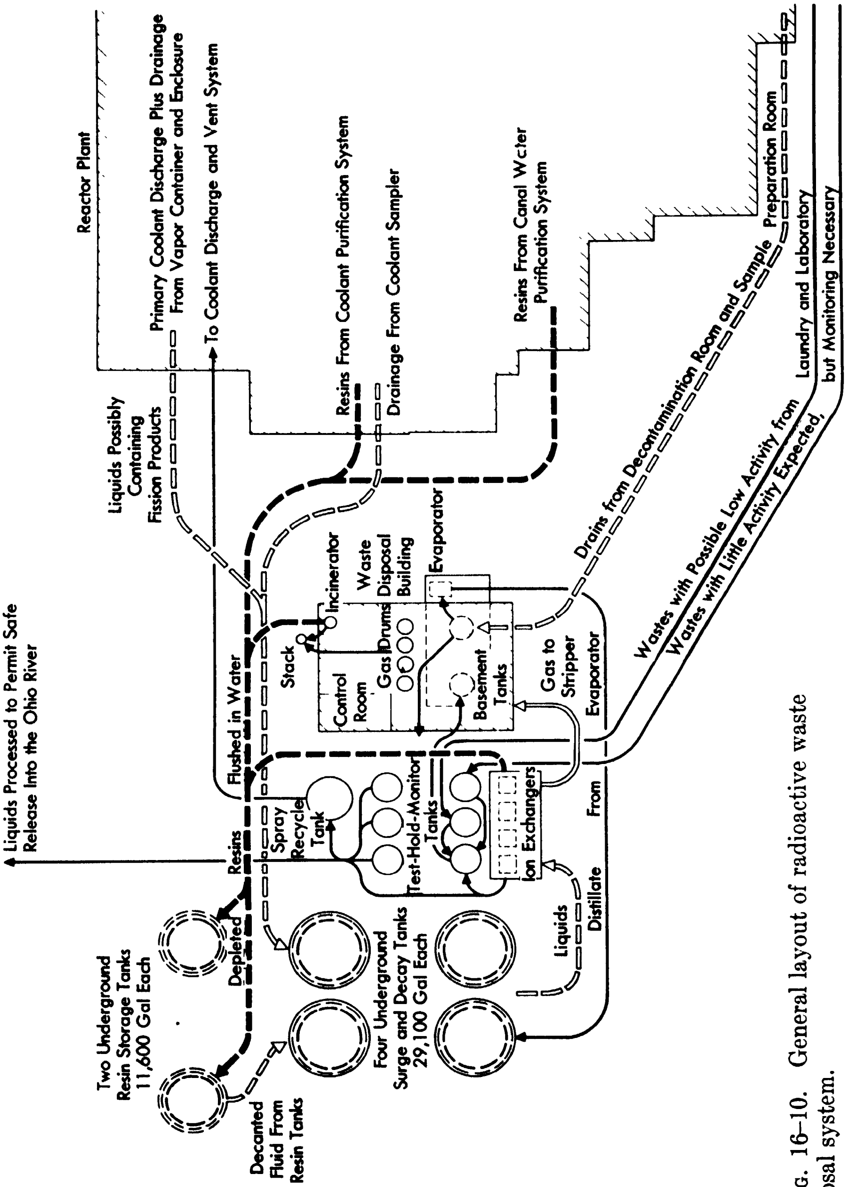


FIG. 16–10. General layout of radioactive waste disposal system.

(2) *Clean tool room* — provides space to store tools and equipment which must be kept scrupulously clean to avoid introducing foreign matter into the reactor coolant system through the equipment assembled in the clean room. This room can also serve as a ready storage area for lint-free clothes and materials used in cleaning.

(3) *Contaminated tool room* for the fuel-handling building—provides space for tools and materials used in the fuel-handling building and the reactor plant container. These tools may be expected to retain varying degrees of radioactive contamination depending on their use, and should not be stored with tools intended for general use.

(4) *Laundry* — located between the clean and contaminated locker rooms. There are two tumblers and two extractors, arranged in pairs. At one end is a work counter for monitoring clothing. That which is "hot" passes through one set of laundry equipment; that which is "clean" is laundered in the other set. At the other end of the room is a counter for folding finished laundry. The laundry is again probed here for radioactivity and if not acceptable is reprocessed.

16-2.7 Waste disposal system structures. The system for disposing of radioactive wastes (Chapter 10) occupies an area of more than an acre adjacent to the main plant. The extensive structural work is estimated to have cost approximately 40% of the total cost of the waste disposal plant; this is much higher than usual for steam power stations and process plants in general.

Some idea of the scope of the structural work may be gathered from the fact that about 5000 cubic yards of concrete were required. Criteria governing the design included:

(1) Tanks and pipes with contents which are or may be radioactive must be adequately shielded and preferably installed underground.

(2) When tanks and pipes are installed underground, the installations must be made so that no hidden leakage can develop which might result in undetected radioactive contamination of the local ground water.

These criteria made it necessary to place these elements in vaults or chambers with space around them for monitoring and checking.

Figure 16-10 shows the general layout of the waste disposal plant and gives some idea of the nature of the installation.

Figure 16-11 shows a cross section through one of the vaults in which the two resin storage tanks are installed. These stainless steel tanks, each with a capacity of 11,600 gals, receive the liquids containing the spent resins from purification of the coolant water in the reactor plant and from the deionization process of the waste disposal plant. These resins may be highly radioactive and will be allowed to settle out in these storage tanks, so that heavy shielding is necessary. Hence, there is an earth cover of

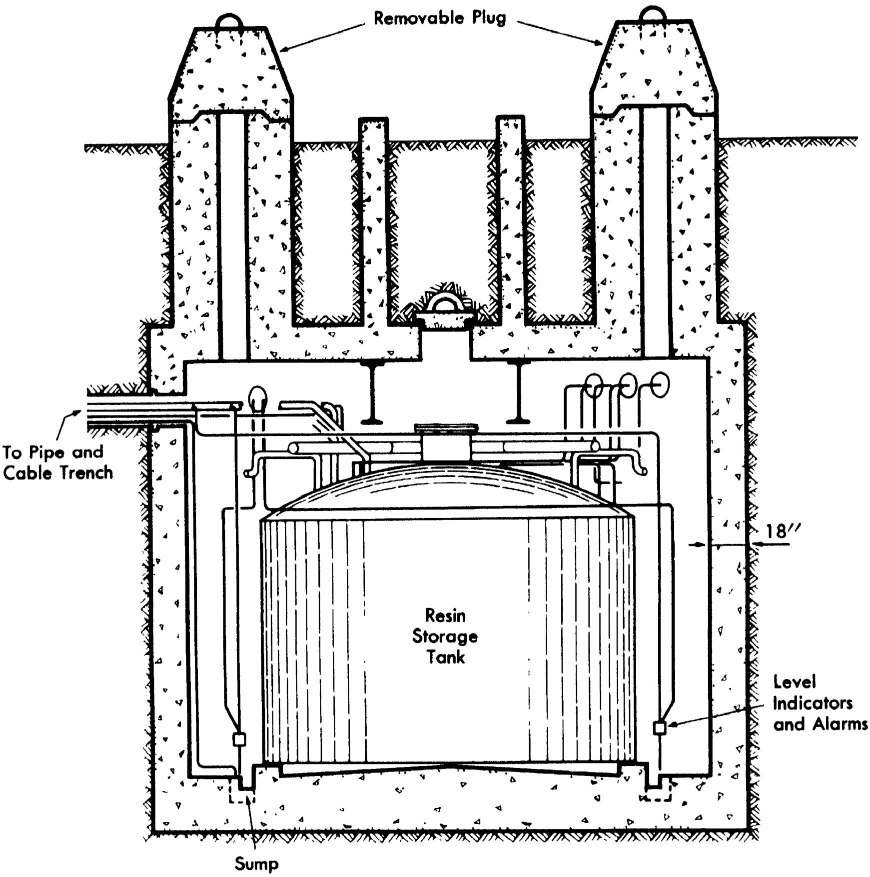


FIG. 16–11. Cross section of resin storage tank enclosure.

$7\frac{1}{2}$ ft above the 18-in. concrete slab over the tanks. There are also four tanks, known as surge and decay tanks, similarly buried, each with a capacity of 29,100 gals. However, since the radioactivity of their contents is lower, only 2 ft of concrete, placed at grade, was required for shielding. Their concrete bases are arranged so that any leakage will drain to the gutters around their perimeters. These gutters lead to sumps in which steam eductors are permanently installed. These sumps are regularly checked, and any water in them is monitored to guard against the remote possibility that it might have come from the tanks or piping instead of being ground water that has penetrated the concrete enclosures.

Access to the enclosures and to the tanks, if necessary and if careful monitoring shows that allowable activity limits for personnel will not be exceeded, is provided. Manholes are provided in the tops of the tanks.

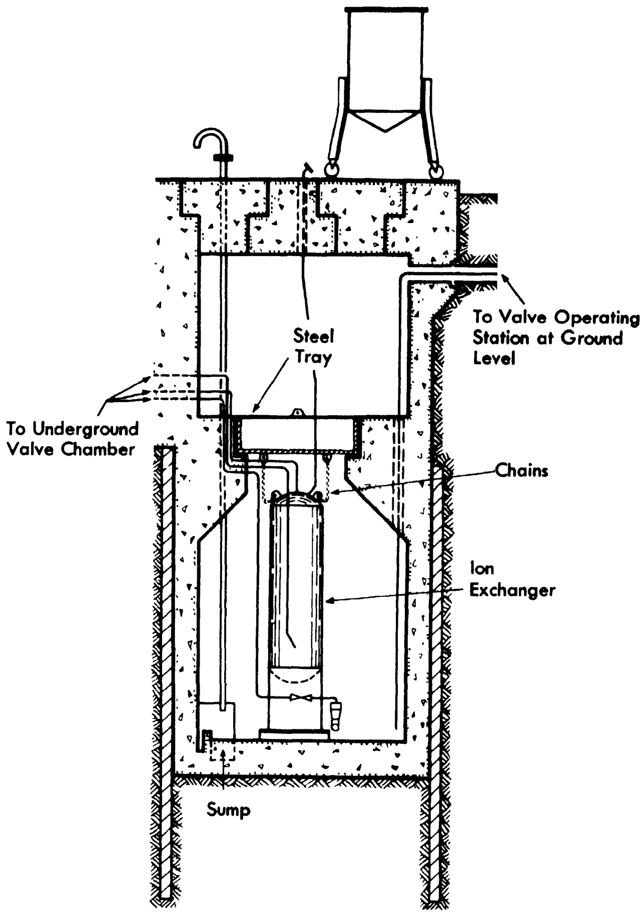


FIG. 16-12. Cross section of ion exchanger vault.

The total cost of the six underground tank enclosures, with the associated access facilities and miscellaneous pits and chambers required, was approximately 30% of the total structural cost of the waste disposal plant.

The foundations of the seven above-ground tanks are so constructed that leakage through their bottoms will find its way out through visible drains where it can be observed and monitored so that corrective measures can be taken if necessary.

The most difficult part of the waste disposal plant design, with regard to radioactivity, was the deionization equipment. This equipment removes radioactivity from the liquid wastes prior to their disposal. There are four ion exchangers, or demineralizers, each comprising a resin-filled stainless steel vessel 2 ft in diameter and 9 ft 6 in. high, placed vertically in a

separate underground concrete vault (see Fig. 16–12). There is a steel tray over each deep vault, above which is a shielded access space. If it becomes necessary to remove a contaminated ion exchanger, this tray can be filled with molten lead or lead pellets poured down through a tube from above. Shielding thus provided makes it possible for a man to enter the access space and disconnect all pipes. The tray can then be hoisted out with a cable extending down from the end of a long boom on a crawler crane. The ion exchanger beneath, which is chained to the underside of the tray with chains and which is held in position only by studs from which it can be lifted, will come up with the tray. The long boom on the crawler crane enables the operator to keep it at a distance, and shielding may be installed around the cab if necessary. The ion exchanger can thus be transported to a nearby burial pit or deposited in a shielded coffin on a flatcar for disposal elsewhere, and a new tray and ion exchanger can then be installed in the vault. The construction of the vault involved deep excavation, and with the removable covers and the adjacent pipe chamber, represents about 12% of the total structural cost of the waste disposal plant.

Most of the rest of the equipment is installed in a one-story building covering an area of 4200 ft^2, the location of which is indicated in Fig. 16–10. A plan of the building is shown in Fig. 16–13. It contains the control room and the principal pumps for moving the various fluids back and forth between the different parts of the plant. It also houses the equipment and gas drums for drawing off and handling the vapors from all the tanks and the gas stripped from processed effluent. There is also in the building an incinerator for disposing of combustible wastes, such as laboratory filter papers, rags used in decontaminating equipment, and the like. Under one end of the building, shielded beneath a concrete slab, is a deep basement in which is installed equipment for the preliminary processing of wastes with a high solids content.

All the various elements of the waste disposal plant described above are connected to a network of piping, carrying liquids which may sometimes contain radioactivity. This extensive network is installed in underground concrete trenches which provide adequate shielding for all eventualities. These trenches have removable concrete covers so that they can be opened up if necessary. The amount of shielding required is provided by earth and concrete combined. In some areas the trenches are in two or three layers.

Where it could be foreseen that excavation to reach a trench at some future time would undermine an adjacent foundation or structure, concrete retaining walls have been built into the trench system to make such excavation feasible. Where electric duct lines have been laid underground close to the trenches in such a manner that they might be undermined, the

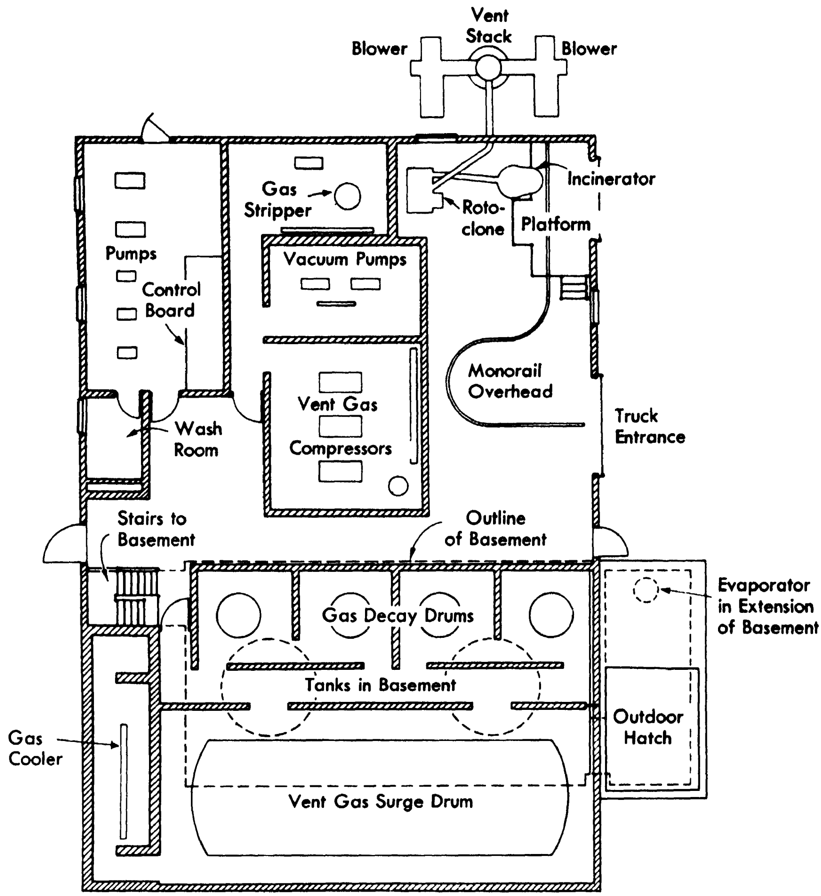


FIG. 16-13. Plan of waste disposal building.

concrete envelopes enclosing the ducts have been mounted on posts which will hold them in position when the trenches are opened.

The trenches are constructed with sampling sumps approximately every 50 ft and are carefully graded to drain into them. These sumps will collect any leakage from the pipes or seepage of water from the ground and can readily be checked through access sleeves. Water, if found, can be monitored for radioactivity. Portable steam eductors can be lowered into each sump through the access sleeves if necessary. The trenches are all laid out so that if the sampling sumps fill up, the overflow eventually runs into the tank enclosure sumps or into a single large central trench sump. The sampling or intermediate sumps at 50-ft intervals will enable the operators to determine by systematic checking the approximate location of leakage.

The subsoil conditions at Shippingport were unusually favorable for constructing the waste disposal plant. The entire waste disposal area is located on a sand and gravel terrace and the yard level is approximately 70 ft above the normal level of the river. Even during the highest floods the river will not rise above the levels of the bottom slabs of the deepest concrete enclosures. The soil is unusually good for foundation purposes and is of such nature that the excavated material could be compacted to form backfill as dense and firm as the original ground. Particular care was taken to compact all backfill, and it is anticipated that there will be no appreciable settlement, so that the trenches laid on the fill will remain close to their original alignment.

Some idea of the extent of the trench construction can be obtained from the fact that the total area of the removable precast concrete trench covers is approximately 12,000 ft^2 and there are approximately 700 separate pieces. Seventeen drawings were required to show the trenches, and the work was estimated to account for approximately 25% of the total structural cost of the waste disposal plant.

16–3. Materials and Finishes

16–3.1 Materials of construction. *Walls.* Except for the enclosure of the reactor plant container and the core vault, which are of reinforced concrete, all buildings are framed with structural steel. Insulated metal exterior wall panels and siding were used on the fuel-handling building. The walls of counting rooms are of reinforced concrete; all other walls and all partitions are of concrete block. The exterior walls are one thickness of 8-in. block and the interior partitions are of 6-in. block. The exterior concrete block walls are coated with a vinyl plastic weatherproofing, sprayed on in five separate coats to give a total dry thickness of 30 to 40 mils.

Roof decking. In the fuel-handling building and the shower rooms, where the underside of the roof could be exposed to high humidity with resultant condensation, the roof decking is of precast concrete slabs. The roof of the core vault and the auxiliary power room are of poured concrete, the former for security and the latter to support power transformers mounted on the roof. All other areas, including the waste disposal building, are covered with metal decking, that over the reactor plant service building actually being a metal subflooring for use if at some time in the future it may be desired to add a second floor.

All roof areas are insulated and covered with a four-ply builtup tar and gravel roofing.

Concrete. All concrete used for structure and shielding is conventional. Both Type I and Type IS portland cement, conforming to ASTM specifi-

cations CI50-153 and C205-53T respectively, was used, the governing factors being cost and availability. Fly ash was used as an admixture replacing 20% of the cement by volume for mixes designed for 28 day strengths up to 3000 psi. An air entraining admixture was used for that concrete which was to be exposed to the weather.

No high density aggregate concrete was used for shielding, since in no case was the space saving sufficient to justify the additional cost.

16-3.2 Flooring and finishes. *Floor covering.* The radiochemistry and cold chemistry laboratories and the darkroom are covered with 1/16-in. vinyl plastic flooring. The offices and corridors, the health physics laboratory, the clean room, and similar areas are covered with a 3/16-in. asphalt tile. The major portion of the decontamination room is covered with stainless steel sheet. Other areas, except for shower floors, in which ceramic tile was used, are painted with an acid resistant plastic (discussed below) with good decontamination characteristics.

Painting. The outside of the plant container is coated with two coats of Bitumastic for a total thickness of 30 mils; the inside, including all internal concrete shielding, miscellaneous iron, and insulation covering, is coated with a three coat vinyl system. In fact, all surfaces within the container are so covered except those of conventional equipment — such as pipe hangers and commercial pumps and valves — which were shipped with a conventional oil base paint. Exposed stainless steel surfaces are left bare. A system of different colors was used to provide contrast for viewing within the container with television equipment. "Dead" areas such as concrete shielding and the container shell are white for best lighting.

A thermal setting plastic was used on the inside of the fuel handling canal, the floors of the fuel-handling building, the walls of the decontamination room, and similar other areas in the reactor plant service building which would be subject to contamination. Some of the walls so covered are of concrete block. A four coat system reinforced with glass fiber was used on the canal.

In this particular case the thermal setting plastic has a distinct economic advantage over the three coat vinyl system, since no special finishing of the concrete surfaces was required. For the vinyl system a smooth surface free of voids is necessary, while the thermal setting plastic fills small voids in the concrete block. This latter material, with the aid of the glass fabric, also spanned voids in the concrete surface up to 3/8 in. in diameter, so that the only finishing required in the canal was to fill surface voids larger than this.

All other surfaces were painted with conventional paints according to general practice.

Supplementary Reading

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2. C. E. Langlois, *PWR Reactor Plant Fuel Handling and Service Buildings; Amplified Facility Design Requirements*, USAEC Report WAPD-PS-332, Westinghouse Atomic Power Division, 1955.

3. Westinghouse Atomic Power Division, *Description of the Shippingport Atomic Power Station* (supplement to the PWR Hazards Summary Report, WAPD-SC-541), USAEC Report WAPD-PWR-970, 1957.

4. Westinghouse Atomic Power Division, Reactor Plant Site Description, App. B, in *Shippingport Atomic Power Station Manual, Volume II*, USAEC Report TID-7020 (Vol. II), 1958.

5. Westinghouse Electric Corporation, *Westinghouse Engr.* **18**(2), (March 1958).

6. H. T. Evans, Structural Features of the Waste Disposal System for the Shippingport Atomic Power Station, Shippingport, Pennsylvania, in *Proceedings of the Second Nuclear Engineering and Science Conference, Vol. 1, Advances in Nuclear Engineering*. New York: Pergamon Press, 1957. (pp. 434–442)

7. R. Lloyd, Decontaminability of Structural Materials and Surface Coatings for Use in Nuclear Installations, in *Bettis Technical Review, Volume I, No. 3, Reactor Chemistry and Plant Materials*, USAEC Report WAPD-BT-3, August 1957. (pp. 84–97).

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CHAPTER 17

CONSTRUCTION

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On Labor Day, September 6, 1954, President Eisenhower started earth-moving equipment at Shippingport by remote control from Denver, Colorado, to open construction of the first large-scale nuclear power plant in the United States. Heavy construction began in May 1955.

17-1. Organization

Under partnership arrangement with the U. S. Atomic Energy Commission, the Duquesne Light Company built the turbine-generator plant and contributed $5,000,000 toward construction of the nuclear plant. This gave rise to a complex administrative situation involving coordination of the work of major subcontractors at the site. The Duquesne Light Company delegated its field work to Burns and Roe, Incorporated, as agent constructors. The Atomic Energy Commission, through its prime contractor, Westinghouse, selected Stone & Webster for architect-engineering and inspection services, and Dravo Corporation as construction contractor to install the nuclear portion of the plant, except for the service building and other superstructures, which were assigned to Crump Incorporated. The Westinghouse Site Activities Manager was responsible for general supervision of all phases of erection, installation, and inspection of the nuclear portion of the plant. He directed the activities of the construction subcontractor (Dravo) and the inspection subcontractor (Stone & Webster).

The Atomic Energy Commission representative at the site coordinated the work of the major AEC contractors, including Westinghouse and the Duquesne Light Company.

Throughout the period of construction a Coordinating Committee, consisting of representatives of the AEC and all major contractors, held regular monthly meetings to coordinate planning and scheduling.

Weekly site meetings with major Westinghouse subcontractors coordinated and expedited the work in the nuclear portion of the plant and constituted the formal contact between the Westinghouse Site Activities Manager and his staff and the Bettis engineering staff. Uncertainties

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arising from the pioneering nature of the endeavor were resolved at these meetings or referred to the proper person for action.

Financial control over the cost-plus-fixed-fee construction subcontract for the nucleai plant was maintained through periodic construction cost estimates, monthly cost and commitment reports, projected manpower estimates, and a constant review of all costs associated with construction by AEC representatives, the Westinghouse Site Activities Manager, and Dravo's project manager. A policy of firm price lower tier subcontracts was followed throughout construction.

17–2. Construction Plan and Features

17–2.1 Planning and scheduling. The Coordinating Committee issued the first master schedule covering all phases of construction in the nuclear plant as well as in the turbine-generator plant. This was necessary because there were many problems associated with manpower, procuring material, access to areas, construction techniques, design, and coordinating the many subcontractors. In March 1956, Dravo was assigned the responsibility of preparing and issuing a master construction schedule for the nuclear plant.

During the late spring and early summer of 1956, the maturing realization of the much greater magnitude of construction resulted in the exertion of extraordinary efforts in planning and scheduling. This was made necessary not only by the adjustment of task magnitude and by the over-all design of the plant, which restricted material flow and confined productive work areas, but also by the relatively untried production techniques which had to be employed.

To bring to the PWR Project the desired experience in planning, scheduling, and constructing a nuclear plant, late in the summer of 1956 the Naval Reactor Facility at Idaho Falls, Idaho, and the Electric Boat Division of General Dynamics Corporation, Groton, Connecticut, furnished personnel who formed the nucleus of the constructor's (Dravo) planning and scheduling group. At the same time, the constructor assigned to this work additional personnel from his own organization.

The expanded group developed over-all construction schedules, based on current knowledge of material and component availability and on unit labor figures for installing the various nuclear plant elements. It was soon found that installation of the piping systems would be the largest single category of work. Of particular significance was the anticipated need for approximately 60 welders qualified to use the consumable insert technique. Advance knowledge of this kind made it possible to conduct the necessary on-site training programs. Thus, enough trained personnel were usually available when required.

From time to time, construction difficulties and delays in component and materials manufacture caused target completion dates to be revised. Such changes made complete schedule revision necessary every few months. Minor revisions in these over-all schedules were required as the result of the determination of weekly progress.

This extensive planning and scheduling of construction work permitted sound manpower planning, timely training programs for special techniques, and advanced spotting of potential construction bottlenecks.

17-2.2 **Progress measurement tools.** Shippingport Station construction work was so diverse and complex that progress could not be assessed visually. The nuclear plant was to be built in four separate steel chambers. Equipment had to be located for maintenance accessibility rather than where pipe runs would be the shortest. These facts introduced more difficulties into assessing the status of work at various construction stages. Consequently, progress measurement tools had to be developed for each major portion of work in the plant.

Each measurement tool was based on some physical yardstick, such as feet of wire or number of welds. Frequent reports and tabulations of units accomplished during the reporting period — usually weekly, sometimes daily — were then required. Total work represented by each area measured was phased into the over-all schedule, indicating the weekly rates necessary to hold the schedule.

Whenever weekly progress fell behind schedule, management could investigate the causes and promptly initiate corrective action. Without the measurement tools, weeks or months might have elapsed before ordinary methods of observing job progress would have brought the deficiency to light. Two specific tools used on this job were: (1) total number of field welds, and (2) total field weld man-hours. Correlated with these was measurement of footage of pipe installed. Approximately 25,000 field welds and some 80,000 ft of piping were involved in the nuclear portion of the plant. Welds ranged from those on 18-in. OD stainless steel piping, with a wall thickness of $1\frac{1}{2}$ in., requiring approximately 150 welding man-hours for one joint, to 1/2-in. carbon steel socket welds requiring half an hour to align and weld. Thus, not only was the number of welds made in each period significant, but the number of man-hours of welding credited, as measured by the number of completed welds of each type multiplied by the unit man-hour estimate for that type of weld, was also important as a trend indicator. In the case of small welds it was found that measurement of the footage of piping installed was a more reliable indicator than either number of welds or number of credited man-hours of welding.

For the electrical work, similar measurements were based on conduit and cable footages, number of terminal blocks to be wired, and number of machines to be connected.

For heavy construction, the commonly accepted practices of measuring progress, such as concrete yardage and structural steel tonnage, were employed.

17-2.3 Stress relieving. Complete thermal stress relief of the pressure vessel was not required. Thermal stress relief was performed, by the use of gas fired rings, only at reinforcements, flanges, and other points required by the applicable codes due to the thickness of metal. Piping was not stress relieved.

17-2.4 Inspection and cleanliness requirements. Since routine inspections practiced in normal power plant construction were inadequate to meet the nuclear plant requirements, special techniques had to be developed. Special training in weld technology and x-ray interpretation was also given inspectors.

All nuclear plant material and components, except standard hardware, were given a thorough receiving inspection. This was necessary, under the extremely tight schedule, to provide sufficient time to reclean, repair, or return units.

Following uncrating and preliminary observation, components which had been shipped in air-tight or pressurized containers were moved to a clean area for inspection. The material was examined for cleanliness, dimensional tolerances, and transit damage. On release by the inspection department, the receiver stored the material or component in a safe location protected from weather and mechanical damage.

Stainless steel pipe was segregated by size and type and stored separate from other pipe. Each piece was marked and color coded over its entire length. Strict warehouse control was exercised at all times.

For prefabricating pipe sections at the site, the following sequence was used: fabrication, cleaning, drying, welding, capping, and storing. All bending was done cold whenever possible. Bending was done before or after cleaning, depending on the suitability of the site cleaning facilities. Stainless steel pipe was cleaned in an alkaline solution by immersing in a solution of 6 to 8 ounces of Oakite No. 19 per gallon of water. The solution was maintained at a temperature between 190 and 210°F. The pipe was immersed for 15 min and in some cases, where heavy grease was encountered, for 30 min. The desired concentration of the solution was maintained by periodically checking with a titrating kit following the degreasing operations. The pipe was thoroughly washed in tap water. The piping received a thorough rinse in unused commercial distilled water containing less than 10 ppm solids.

A clean area was required for stainless steel welding. The over-all plant container could not be maintained as a clean area because other work, such as concrete and equipment setting, was carried on at the same time. Therefore it was necessary to make the individual welds in the 18-in. piping of the reactor coolant system inside temporary shelters constructed within the reactor plant container and which were supplied with clean air. The welders, helpers, and fitters were required to wear lint-free white coveralls, gloves, and caps. After welding a section of piping, the pipe systems were capped to prevent dirt entering. A colored tape system of inspection identification was devised to insure that all pipe had been inspected prior to closure welding. Before making final closure welds of the auxiliary systems into the reactor coolant system, all lines were flushed with distilled, demineralized water to remove dirt, weld spatter, and other foreign particles.

Butt welded joints in both the type-304 stainless steel pipe and the carbon steel pipe which are subject to fatigue or impact were inspected as follows:

(1) Dye penetrant test of root weld.
(2) Dye penetrant test of final weld.
(3) Radiographic test of root weld on main coolant and auxiliary system stainless steel piping.
(4) Radiographic test after completion of 3/8-in. weld throat (18-in. reactor coolant pipe only).
(5) Radiographic test of final weld.

Radiographic examination of reactor coolant system piping utilized a 20-curie iridium source; of 6-in. pipe, a 4 to 5-curie source; and of 2-in. pipe, a 2 to 4-curie source.

17–2.5 Pipe welding and welders' training. The 18-in. diameter reactor coolant system pipe, as well as the reactor coolant auxiliary piping, all of type-304 stainless steel, was welded by means of consumable weld inserts similar in composition to the base metal. The root pass for the weld insert was performed by use of a tungsten arc process, and the internal area of the joint was purged with argon gas. The remainder of the weld was then made by the metal arc process. An escape opening was provided in the bulkhead opposite the inlet for the gas to permit displacement of atmospheric gases. The purging gas flow was limited to a maximum of 20 cfm. No stress relieving or preheating was required for any stainless steel pipe. Joint identification record cards were kept on all stainless steel pipe joints in the coolant systems, thus identifying material specifications, supplier, mill certifications, welder, and inspector.

The welding technique for the reactor coolant system piping was completely foreign to otherwise qualified welders, making it necessary to set up training facilities at the site. Experienced instructors were obtained

from other nuclear installations. The training period averaged approximately three weeks before the welders qualified under established welding procedures. Progress after qualification was at first slow because the men realized they were dealing with expensive, difficult-to-replace material and proceeded with extreme caution. The requirements for cleanliness introduced a new factor which was unique in the experience of the men doing the work. Initially, the difficulties encountered were due to slag and tungsten inclusion, porosity, and burn-through. As more knowledge was gained with the completion of each segment of these loops, progress was much more rapid. The standard of workmanship finally achieved was extremely high, proven by the fact that subsequent hydrostatic tests at 150% of design pressure revealed not a single butt weld leak. This included the auxiliary system stainless steel piping using a similar welding procedure.

17–2.6 Area and system work. The construction drawings for this plant were developed on an area basis so that all piping for one area was shown on the drawing or the series of drawings that represented that particular area. Twenty-four separate areas were utilized. With drawings by areas, a construction crew could use a minimum number of drawings to cover all the work in one physical location, thereby increasing construction efficiency. This mode of operation was effective through approximately the first 80% of construction. At this point, however, partial testing of completed systems began, and completion schedules were required on the basis of systems rather than areas. As a result, several check lists were developed, each list showing each specific run of pipe and cable, as well as equipment and instrumentation required for the completion of the particular system or portion of a system to enable it to be tested. About 50 such check lists were developed when construction was approximately 80% complete. When construction was about 90% complete, almost all of the forces worked in accordance with check lists rather than on an area basis. In certain instances job efficiency demanded that part of the work still be carried out on an area basis.

17–2.7 Reactor plant instruments—non-nuclear. One-inch, schedule 80 stainless steel pipe was used to connect the pressure and differential pressure transmitters to the system piping, with capped manually operated valves serving as block and bypass valves. The transmitters were welded to these pipes; temporary sensing elements were welded directly into the system piping for test purposes. Electric transmission is used exclusively to transmit the intelligence from the transmitters to remote receivers in the control room. The receivers use static mechanical amplifiers to power the servomotor-driven units. The receivers provide indication of

control or alarm functions, and retransmit to recording or remote indication units. All instrumentation was checked by impressing a simulated input on the transmitter and leading the signal out of the receivers into the control room. This method insured that the instrument, instrument piping, and instrument wiring were correct.

17-2.8 Temporary facilities. Temporary facilities required for the construction of the nuclear portion of the plant provided work space for 720 men, the peak manpower reached on day shift during construction. A number of field shops were required to facilitate field fabrication of concrete forms, miscellaneous structural steel, miscellaneous pipe hangers, and stainless steel pipe two inches and smaller. The major portion of larger stainless steel pipe was fabricated off-site.

The rigid pipe cleaning requirements called for installing extensive fabrication and cleaning facilities. The alkaline solution wash was done in a steel tank 22 ft long, 6 ft wide, and 4 ft deep. The cleaning solution was heated by a portable 34.5 hp steam generator with an equivalent evaporation of 1190.25 lb of water per hour. The clean rinse was done in a steel tank 22 ft long, 6 ft wide, and 30 in. deep. Water softening was done by means of five 50,000 grain service softeners, which were recharged as required.

Because all butt welds in the stainless steel pipe had to be x-rayed, a special room in one end of the pipe-fabricating shop was set up for this purpose.

17-3. Special Construction Techniques

17-3.1 Equipment and material handling. The two boiler chambers are each 50 ft in diameter and 97 ft long, and are equipped with two 10-ft diameter overhead access hatches for entry of all equipment and material. The auxiliary chamber is 50 ft in diameter and 147 ft long and was serviced by a temporary air-driven construction hoist which handled all material and equipment up to five tons. All heavier equipment in the auxiliary chamber is located directly below the access hatches; it was handled by the 125-ton overhead crane. The reactor chamber is located directly under the main crane and presented no problem in equipment handling. When major components were being installed, the two boiler chambers were serviced by two 50-ton stiff leg derricks. The two 10-ft diameter boiler chamber hatches were also serviced by a 15-ton monorail hoist, which is part of the permanent plant equipment.

Among the larger equipment installed were the heat exchangers and steam drums. The steam drums, weighing approximately 28 tons, were approximately 27 ft long by 5 ft 6 in. in diameter. Two of the heat ex-

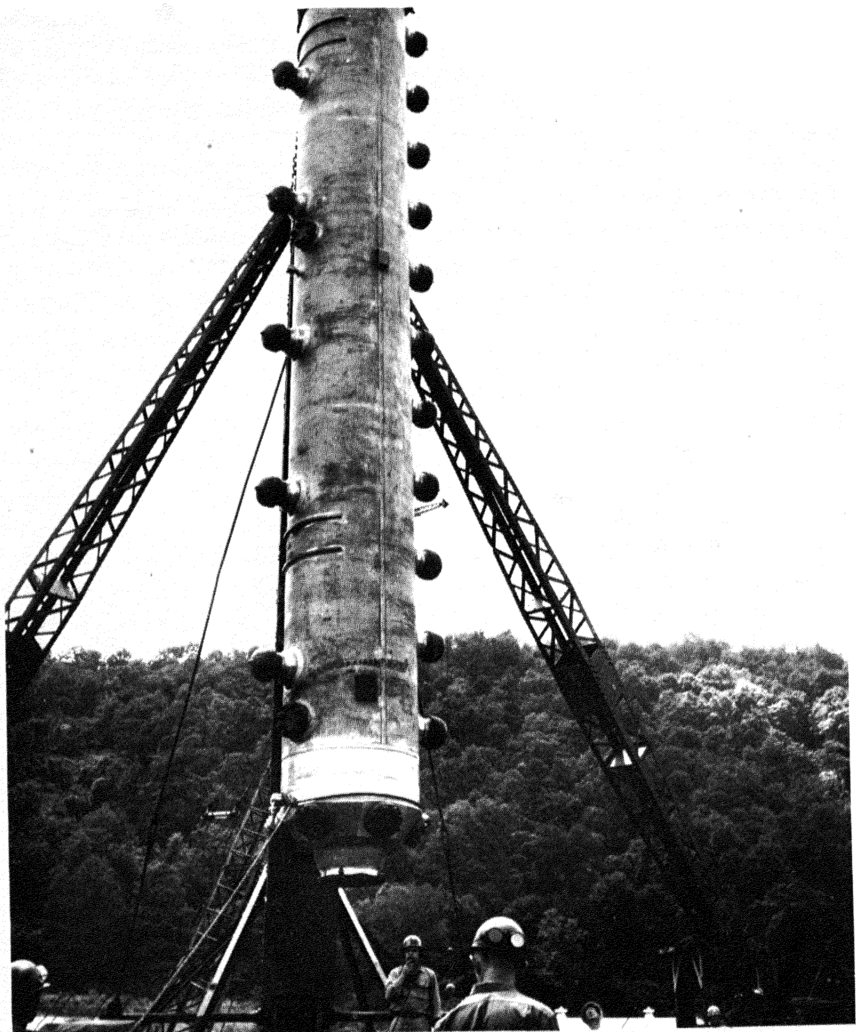

FIG. 17-1. Foster Wheeler heat exchanger being lowered into boiler chamber.

changers, 36 ft long by 3 ft 6 in. in diameter, weighed approximately 29 tons. The U-tube design heat exchangers had an over-all width of 9 ft 3 in. and an over-all length of approximately 29 ft, weighing approximately 32 tons. To test the complex rigging required because of the restricted access, a trial run at 125% of the actual weight to be lifted was conducted using a tank from a railroad car, filled with sand, to simulate the approximate size and weight of the components to be handled. This tank, as were the final plant components later, was lowered through one of the hatches in a

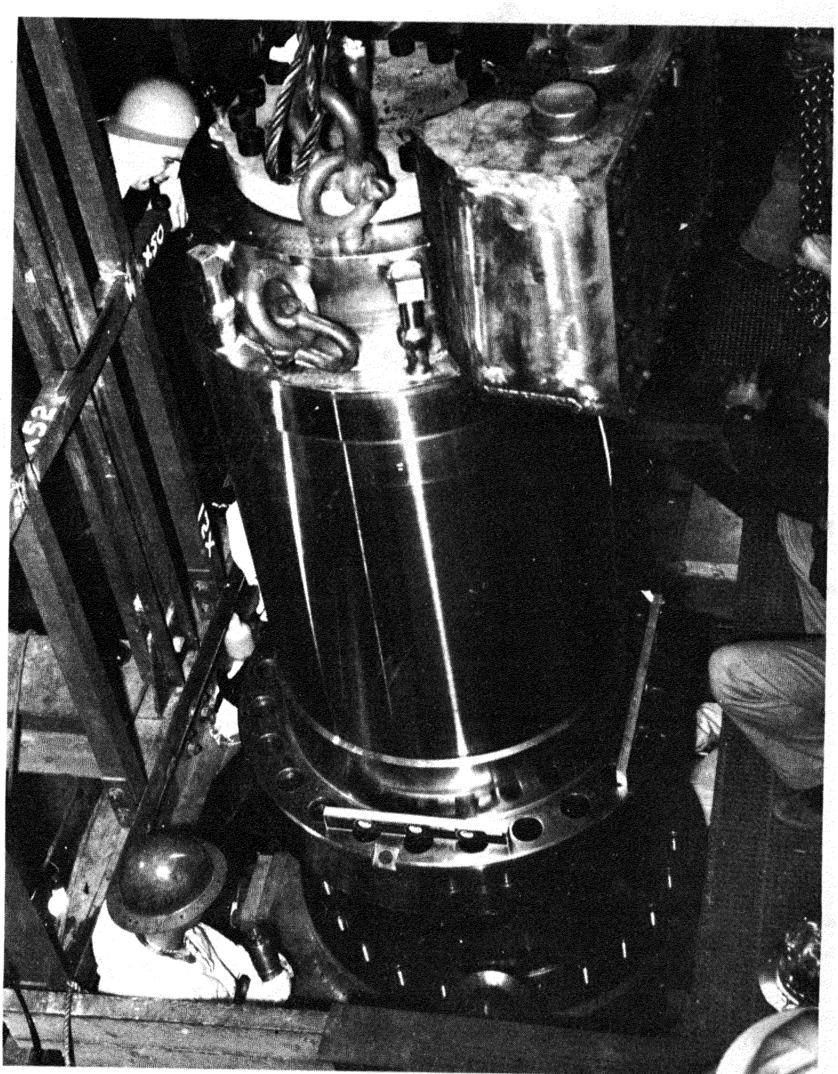

FIG. 17-2. Main coolant pump being lowered into position.

vertical position and then, through the use of a hoist operating from the other end of the chamber, was maneuvered to a horizontal position and placed in its proper location. The trial run was completed by removing the tank. Immediately afterward the steam drums and heat exchangers were installed in like fashion (Fig. 17-1).

Inside three of the four chambers of the reactor plant container, concrete shielding walls were installed. Ready-mix concrete trucks, using pumping

equipment, supplied a total of approximately 2800 cubic yards of concrete for these walls.

Many pieces of equipment smaller than the steam drums and heat exchangers were placed inside the container through the access hatches (see Fig. 17–2) along with many miles of piping ranging from 1/2-in. diameter up to 18-in. diameter — both in fabricated sections and straight lengths. Similarly, all electrical conduit, wiring, and equipment passed through these hatches.

Much temporary material and equipment was required during the construction phase. This included welding machines, temporary heating equipment, temporary power and lighting, switchboards and lines, and an unusually large amount of scaffolding material. The large requirement for scaffolding material resulted from the location of equipment, conduit runs, pipe lines, etc., at varying elevations both within the chambers and in the area between the steel chambers and the concrete enclosures. Temporary equipment and material were brought in and removed from the area between the chambers and the enclosures by temporary air driven construction hoists located at convenient openings in the enclosures.

17–3.2 Construction of reactor plant container and secondary shield. The first concrete, a bottom slab which had the dual function of providing a base and a barrier outside the steel reactor plant container, was placed on August 16, 1955. The bottom concrete consisted of about 10,000 cubic yards of heavy slabs, 7 ft thick except under the reactor where they were 5 ft thick, and a 5-ft thick wall about 15 ft high. This work was completed by November 15, 1955. By carefully scheduling the sequence of bottom concrete pours, Burns and Roe was able to start the canal walls late in September 1955 and to complete them to their final elevation at about the same time as the whole bottom concrete. The canal walls then gave the steel erector a platform upon which to erect derricks with which he could build the steel chambers. These walls were designed so that the area where the eight interconnectors between the reactor and boiler chambers penetrate the concrete could be left open. This allowed working space for the steel erector. For the same reason, the floor of the canal over the reactor chamber was not placed until almost a year later.

The task of fabricating, erecting, and testing the huge (600,000 cubic foot) reactor plant container was awarded on a firm price basis to Pittsburgh-Des Moines Steel Company in April 1955. Erection began in November of 1955 and was scheduled for completion by April 15, 1956. The job was very difficult and the schedule extremely tight. Until the container was tested, little else could be achieved. Through the winter of 1955–56 working conditions were far from ideal for field welding and progress was slow. The erector chose to use two 50-ton derricks erected

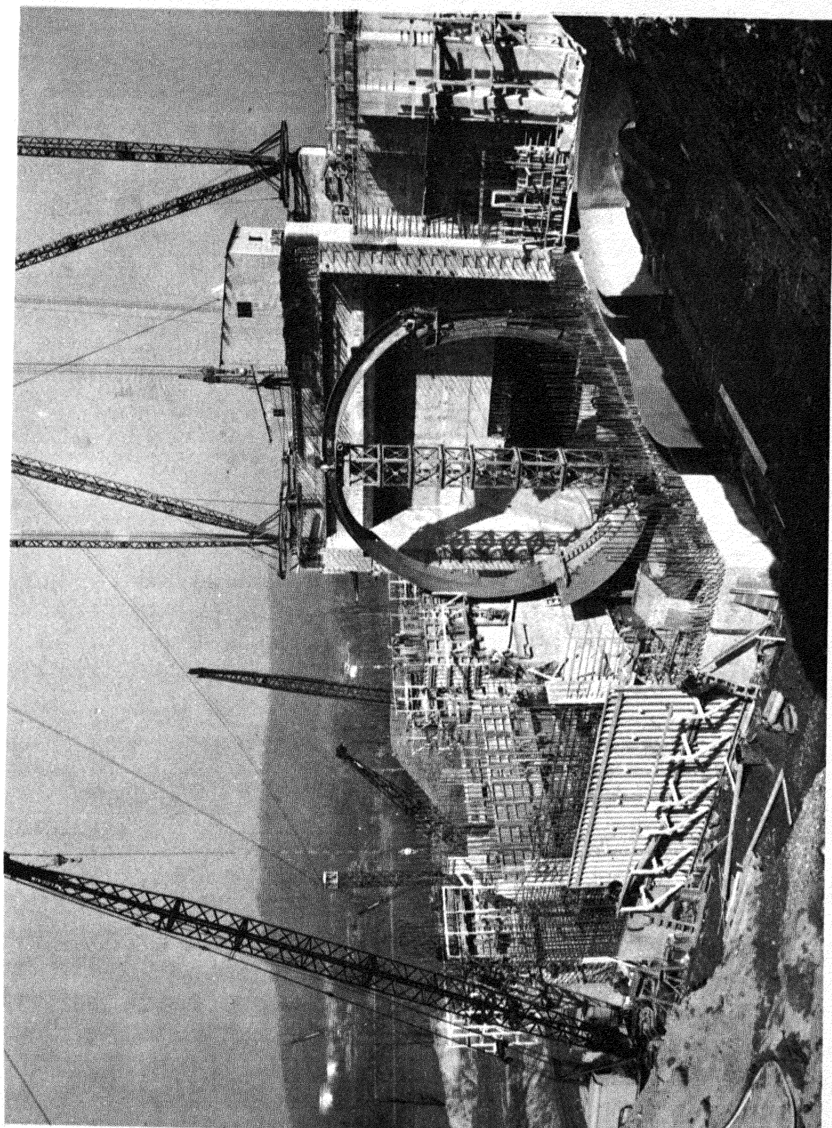

FIG. 17–3. Erection of west boiler chamber.

FIG. 17-4. Aerial view of plant on April 19, 1956, showing boiler and auxiliary chambers and canal walls.

on the walls of the canal. This large size was dictated by the approximate 100-ft distance to be spanned for picking up plate (Fig. 17-3). The size of the job is best illustrated by the fact that the erector had to place and weld more than 2200 tons of steel, performing over 13,000 ft of $1\frac{1}{4}$-in. thick field welding which required 100% x-ray inspection. The quality of the welding was so high that less than 1% required repair. Figure 17-4 shows progress of the work on April 19, 1956.

This extremely difficult welding job, complicated by winter weather and necessary design changes, delayed completion and testing of the container until September 1, 1956.

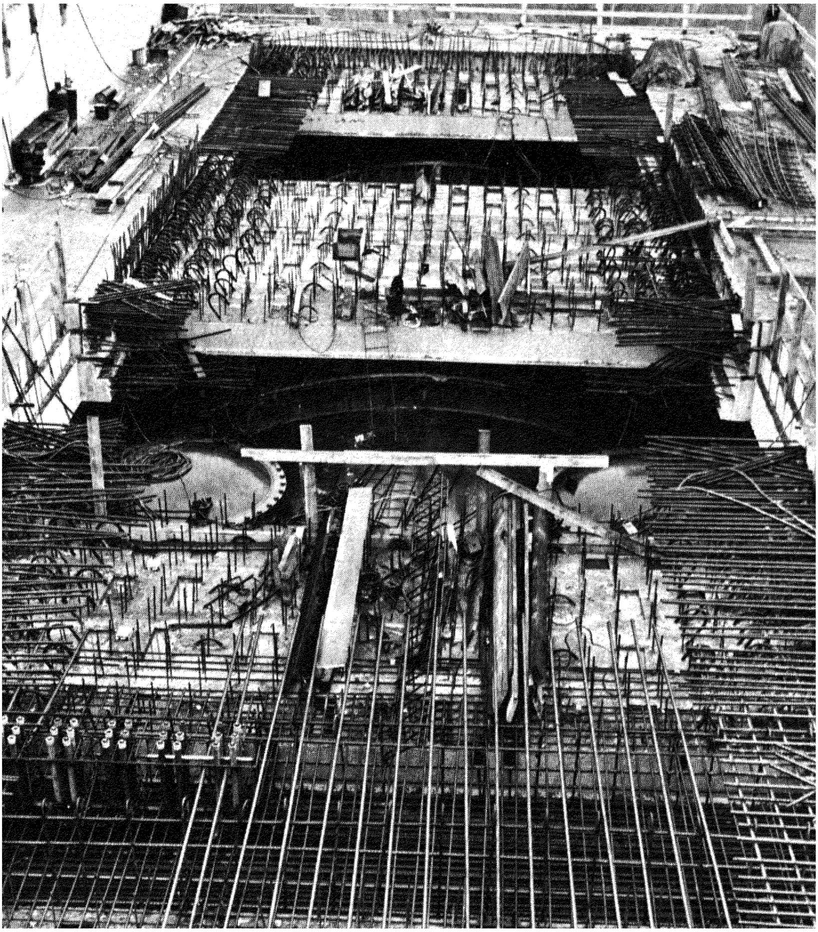

FIG. 17-5. Top deck of east boiler chamber during construction.

17-3.3 Top deck construction. Originally, plans were to leave placing of the top concrete until the vapor chambers were erected and tested. Early in 1956 it became obvious that the steel erector would not meet his target date. It was decided to place the concrete walls and roof.

The straight walls were relatively easy to form and place; the most serious restriction was the three days' curing time required between adjacent pours. This precaution was taken to ensure complete shrinkage within a pour, thereby producing a crack free wall and roof.

A more difficult problem was the haunch transition between the wall and the top deck, necessary to provide a support for the 10-ft cantilevered

section. This was solved in two ways. Sections parallel to the longitudinal axes of the chambers were supported on the barrel section ribs designed for this purpose. The corner sections were supported from the lower concrete deck, approximately 55 ft below. When the haunch pours were completed for both sides of a given deck area, it was possible to form and place the center sections. These were of two types — poured-in-place and precast. The design for those poured in place required heavy wide flange beams over 50 ft in length to act as a strongback and support the center section from above. Because steel of this type was expensive and hard to locate, it was decided to use Bailey bridge sections in lieu of steel beams. Bailey bridge material was obtained from the U. S. Army Corps of Engineers' Marion Depot.

Seven poured-in-place sections and eight precast sections were arranged alternately in a north and south direction to cover the entire structure (Fig. 17–5). The precast sections could be placed only after both of the adjacent poured-in-place sections had been formed, placed, and stripped. They consisted of a series of precast beams 2 and 4 ft wide and covering 2/5 the depth. Thus the precast sections, when placed and grouted, served not only as structural members, but acted also as a form for the upper 3 ft of concrete. The final pours over the precast sections were quartered and opposite quadrants were placed and cured prior to the final two quarters. This reduced to an absolute minimum the possibility of shrinkage cracks.

17–3.4 Fuel-handling canal. To construct a concrete canal of the required size and shape, free of cracks and seepage, demanded extraordinary precautions. All drains were installed above the Ohio River flood level to prevent ground water from accumulating behind walls, which would add hydraulic pressure to normal earth pressure. Backfilling against the east and west ends of the vapor container enclosure was delayed until internal buttress work was complete. Backfilling against the south walls of the enclosure presented many problems because the area was the main access for auxiliary vessel erections, superstructure erection, canal concrete pouring, and equipment and material placement. The architect-engineer advised against backfilling here until the concrete enclosure was wholly completed and had cured enough to develop the rigidity and stiffness designed into it. It was reasoned that the fill compacted against the enclosure, plus the added pressure of heavy construction equipment operating on top of the fill, would tend to deflect the tied-in enclosure, causing a strain on the canal structure that (in spite of heavy reinforcement) might produce cracks most difficult to repair. Planning showed that this area was fast becoming a bottleneck. Drastic action had to be taken to prevent the construction program from bogging down. Therefore, concrete

cribbing was installed to permit backfilling and compacting of fill to make the area available for construction work on the main portion of the reactor plant.

The canal floor over the reactor chamber, a concrete slab 5 ft thick, was installed by utilizing intrusion grout in forms suspended from the fuel-handling operating floor above. The slab at this point supports water 24 ft 6 in. deep, as well as the massive interior partitions of the canal. Although the grout was pumped into the area with extreme care and under the supervision of specialists, a number of surface voids were found after stripping of forms. These were repaired by chipping out the surface concrete uniformly and waterproofing with Ironite.

Placing the canal gates that divide the fuel-handling canal and the four compartments was completely reversed from normal procedure. Ordinarily the concrete walls supporting the gates would be poured first; then the gates would be installed. However, to ensure that they were absolutely watertight when in operation, the gates were first erected in the guides and then the walls were poured in the correct position. The gates, two of ten tons and two of fourteen tons gross weight, allow closing off the water from particular areas of the canal as operating conditions require. The completed canal structure was given a coat of thermosetting plastic waterproofing.

17-3.5 Pipe repairs. The stringent welding procedures and weld inspections followed for stainless steel piping, and in some cases for carbon steel, resulted in repairs considered unnecessary in normal power plant practice. The problems encountered in stainless steel butt welding were predominantly (1) oxidation of the consumable insert due to loss of purge and (2) burn-through. These items accounted for approximately 95% of all repairs, tungsten and slag inclusions for the remainder. Where oxidation was encountered, the affected area was ground out when accessible. Burn-through required localized grinding of the affected area and buildup by coated rod. In more serious cases the joint was ground open, the ends again prepared, and the joint rewelded.

On completion of the closure weld of one 18-in. loop, oxidation and a very slight burn-through occurred but did not show on the x-ray film until the completed weld was x-rayed. Boroscopic inspection revealed the deficiency to be beyond the hydraulic valve with a minimum internal diameter of 13 in. The average welder was too large to enter beyond the valve, and cutting into the system would have been quite expensive. A midget qualified welder was brought to the site. He entered the pipe from the reactor vessel and made the repairs.

Slag inclusion was the only welding problem encountered in socket welding stainless steel piping. Welding carbon steel pipe brought about

the usual porosity and slag inclusion. These were corrected by grinding away the fault and welding.

The modified butt welds used in the main steam, boiler feed, and safety injection lines were very difficult to repair. This particular butt preparation has a machined and rolled lip. When burn-through of any significant magnitude occurred, the joint had to be prepared again, requiring considerable time and effort.

The open butt weld, utilized on carbon steel pipe where a backing ring is a potential trap for radioactive contamination, caused a high rate of rejection because of burn-through. Only after considerable experimenting was it possible to qualify a limited number of welders for this particular type joint.

17–3.6 Loop closure pieces. The need for specially machined closure pieces in the stainless steel reactor coolant system piping was based on three factors:

(1) Minor deviation between drawing location and actual location of equipment.
(2) The amount of distortion experienced with the type of material being installed — type-304 stainless steel.
(3) The joint design, which specified a fitting tolerance of 1/32 in. in order to assure sound welding.

It was decided that two closure pieces in each loop would be necessary to solve the problem. It was further determined that these closure pieces must be as close to the reactor vessel as possible, since the location of the vessel was fixed. Eight such installations had to be made by first fabricating a template at the site. This template consisted of two stainless steel rings matching the J bevels on the sections of pipe to be joined. The rings were connected by means of 8-in. structural channels, thus making a rigid assembly with the exact angularity and dimensions required for the final closure sections of the 18-in. pipe. Since the entire assembly was subject to 3/16-in. shrinkage during the welding process, the closure pieces had to be made that much larger to compensate for the shrinking.

The final closure pieces were then machined by the piping fabricator to the exact dimension of the mock-up. When the closure pieces were installed, the entire coolant loops, all pumps and valves, the heat exchanger, and the boiler drum had to be moved back to make room for the oversize closure pipe. Later shrinkage during the welding then drew them all back into proper position. The huge suspension systems which carry the heat exchanger and boiler drums are designed to permit movement resulting from expansion of connecting pipe lines when the plant is operating. This flexibility allowed for jacking during construction so that final connections could be made. The method resulted in fit-ups of closure pipe joints well

within the specified tolerance, making further field grinding and welding preparation unnecessary. As a consequence, the closure welds were of uniformly high quality and completely acceptable.

17-3.7 Vessel handling and neutron shield tank. The installation of the reactor vessel and the neutron shield tank inside the 36-ft 6-in. ID sphere with an 18-ft ID top opening presented some unusual rigging problems. The reactor vessel is a right cylinder with one end a hemisphere and the other end open but flanged to receive a hemispherical dome. As received from the manufacturer, without the final closure head, the vessel had an over-all length of 25 ft and weighed 153 tons.

The combination neutron shield tank and reactor vessel support were too large to be shop fabricated and shipped to the site. Therefore, shop trial fitting and field erection were necessary. When completed in the shop, the reactor vessel support was shipped complete as one piece and welded into the reactor chamber. All sections of the neutron shield tank inner wall were then placed in the reactor chamber and lashed to the sides of the chamber until ready for assembly. The problem then was to transfer the reactor vessel from its horizontal position on an oversize flatcar, upend it to a vertical position, and suspend it thus in the chamber, approximately 5 ft above its permanent position, while its thermal insulation and supports were fitted. It would then be lowered to its final position by means of jacks. Because of the extraordinarily heavy weight of the vessel, the limited space in which it had to be maneuvered, and the peculiarities of its design, the rigging problems encountered in this operation were unique and complex.

Since the vessel had no lifting attachments, the only points of support for transferring it from the flatcar to the chamber were on the bolt flange. Special lifting lugs, designed to fit the bolt holes in the flange, were pinned to a lifting beam suspended from the hook of the overhead 125-ton building crane. Each component of the lifting beam was tested for shear, torsion, and tension at 125% of the final lift. With the approval of the crane manufacturer, the crane was then tested at 165 tons. When the special flatcar arrived at the site, an oak cradle was installed under the spherical end of the vessel. To prevent the cradle from sliding, angles were welded to previously installed 2-in. stiffener plates. The vessel was upended from its horizontal shipping position on the flatcar and picked up by the crane. It was then transported 100 ft on the crane runway to the centerline of the chamber and lowered 55 ft into the chamber (Fig. 17-6).

It was impractical, because of the construction schedule, to use the crane for supporting the vessel in its suspended position for any length of time. Therefore, a suspension frame was erected across the top of the fuel handling canal. To this frame was attached a system of eyebolts and equalizing slings which were then fastened to the vessel flange at quarter points. The

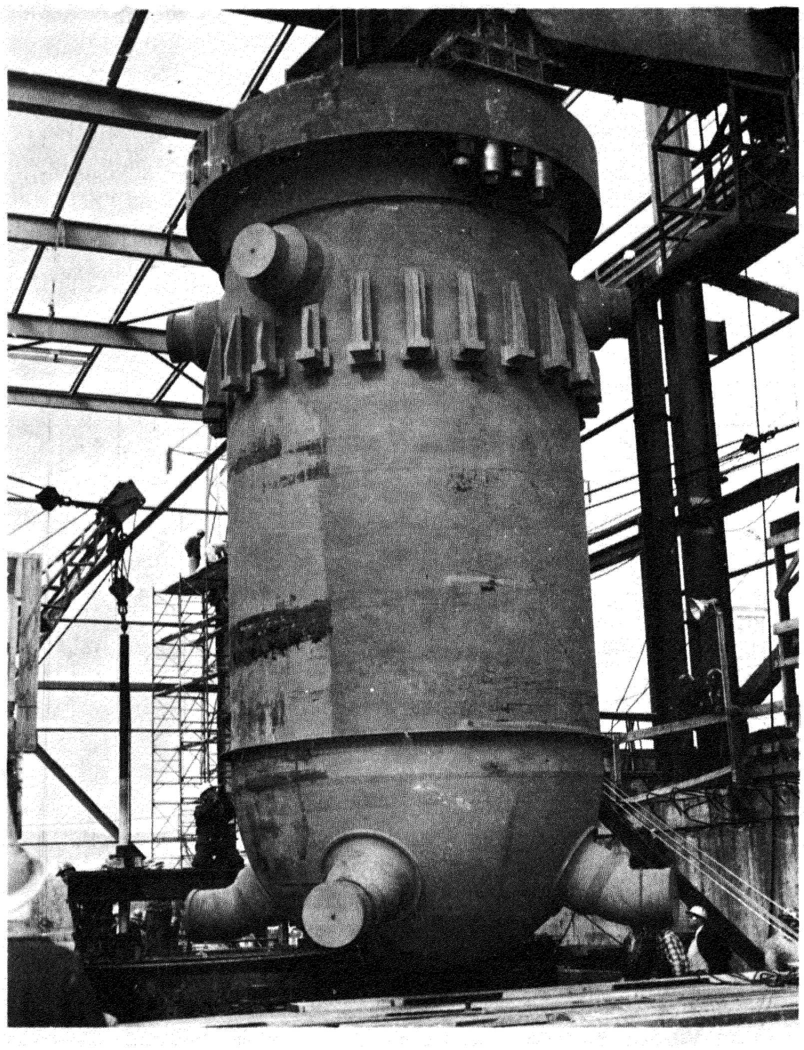

FIG. 17-6. Lowering reactor pressure vessel into reactor chamber.

vessel was lowered enough to transfer the weight from the crane to the suspension frame. The vessel remained in this position from October 1956 until February 1957, while thermal insulation and stainless steel cladding were applied. To assure equal loading at each point of suspension, periodic strain gage readings were taken and compensations made as necessary.

The vessel was lowered to its final position by means of jacks on top the suspension frame. The eyebolts were threaded; two sets of nuts on these bolts were used for down-jacking. Once the vessel was in place, the

suspension system was discarded and the vessel made ready to receive its component parts.

17-4. Integration of Construction and Testing

Early in 1957 a stage was approached in construction where the responsibility for particular plant systems passed from site construction forces to a test and operations group. It had been decided to transfer the various systems to the latter group progressively as they were completed. A Test Support Group, which issued daily bulletins noting system deficiencies, was activated. These deficiencies were given highest priority by the construction forces. To assist the test and operations personnel, construction groups were placed on a shift basis. Daily meetings were held to integrate construction and test work. On October 6, 1957, the reactor core was installed, and construction was considered essentially complete.

17-5. Construction Statistics

Table 17-1 lists various quantities in connection with construction of the nuclear portion of the plant and the radioactive waste disposal system.

TABLE 17-1

PWR CONSTRUCTION QUANTITIES

Concrete		
Concrete enclosure and fuel-handling canal		24,000 yd^3
Service building		1,823 yd^3
Internal shielding		2,740 yd^3
Waste disposal		5,148 yd^3
	Total	33,711 yd^3
Reinforcing		
Concrete enclosure and fuel-handling canal		2,840 tons
Service building		100 tons
Internal shielding		200 tons
Waste disposal		200 tons
	Total	3,340 tons
Structural steel		
Fuel-handling building		460 tons
Service building		90 tons
Waste disposal		33 tons
	Total	583 tons
Miscellaneous steel (supports, walkways, hangers, etc.)		
Reactor plant		700 tons
Fuel-handling building		10 tons
Service building		80 tons
Waste disposal		10 tons
	Total	800 tons
Other steel		
Reactor plant container		2,200 tons
Reactor plant vessels and tanks		640 tons
Sheet piling		800 tons
Boilers and heat exchangers		300 tons
Canal gates		25 tons
Canal water seals		24 tons
	Total	3,989 tons
Pipe		
Carbon steel:		
Nuclear plant		28,707 ft
Waste disposal		18,161 ft
		46,868 ft

TABLE 17-1 (*continued*)

Stainless steel:		
Nuclear plant		22,763 ft
Waste disposal		6,636 ft
		29,399 ft
Others		4,732 ft
	Total	81,000 ft
Valves		
Carbon steel:		
Nuclear plant		440
Service building		64
Waste disposal		810
Stainless steel:		
Nuclear plant		1,320
Service building		24
Waste disposal		162
Welds		
Nuclear plant		15,500
Waste disposal		10,523
Cable		
Nuclear plant (including waste disposal)		582,758 ft
Service building		84,507 ft
	Total	667,265 ft
Conduit		
Nuclear plant		143,460 ft
Concrete block		
Nuclear plant		28,470 ft^2
Roofing		
Nuclear plant		35,300 ft^2
Wall panels (external)		
Nuclear plant		32,500 ft^2
Motors		
Reactor plant (including waste disposal):		
Number		173
Total horsepower		9,673

CHAPTER 18

TEST PROGRAM

CHAPTER 18

TEST PROGRAM*

18-1. Introduction

The test program for the Shippingport plant was intended to accomplish three things: (1) ensure that the plant was thoroughly checked out so that it could be started up and operated safely, (2) show that it met its design requirements, and (3) gain information useful not only in this plant but also for designing and operating future plants.

Testing required to permit safe startup and operation was completed a few days before the turbine generator was synchronized. Tests required to demonstrate that the plant meets its design requirements are essentially complete, as of this writing (June, 1958), with no important deficiencies discovered to date. Testing to gain information for designing future plants has only begun; it will continue for several years. This chapter will cover in detail only the testing already completed, with a brief outline of further testing planned for the future.

One of the requirements imposed on the testing to meet the first objective—safe startup and operation—was that it be carried out rapidly. This was satisfactorily met: 4½ months elapsed between physical completion of the reactor coolant piping system and criticality. Two months were actually down time for installing the reactor core and instrumentation. Only 21 days were required between initial criticality and full power. The very short period between initial criticality and full power was made possible only by using information gained from an earlier full-scale critical experiment performed at Bettis Plant, and extensive hydraulic and heat-transfer tests at Bettis and elsewhere on sample fuel elements and core mockups. Thus most tests on the reactor were simple confirming tests rather than detailed investigations. Consequently, few surprises were encountered during the startup, and none required extensive investigation before proceeding with the planned program.

A second factor that helped speed reactor startup was the extensive preliminary testing on parts of the plant as they were completed, culminating in a test of the entire reactor plant at design pressure and temperature before the core was inserted. Most deficiencies that would have caused delay had already been located and corrected before reactor

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startup. Since the reactor core was not available for insertion until completion of the test of the reactor plant at design pressure and temperature, these preliminary tests saved considerable time.

18–2. The Program

18–2.1 Planning the program. After the conceptual plant analysis was completed and the plant parameters were established, the test program was organized late in 1954. Two important facts taken into account in the planning were:

(1) It was apparent quite early that the reactor core would probably be the last important plant component to arrive at the site. It was, therefore, clear that if the plant was to make power on schedule, it would be necessary to reduce to a minimum the testing time from core insertion to full power.

(2) It had already been decided that before the core could be inserted in the reactor pressure vessel, the reactor plant must be operated at normal operating pressure and temperature with a filter in place of the core, to ensure that the reactor coolant systems were dirt free and properly conditioned chemically. This would prevent large particles from clogging the core passages or small particles from coating their heat-transfer surfaces as well.

These requirements resulted in the following decisions:

(1) The filter for installation in the pressure vessel would be made with a hydraulic resistance equal to that of the core. The period during which the system was to be flushed would also be used to complete as much testing and trouble-shooting of the reactor plant systems as possible.

(2) The characteristics of the reactor would be determined by use of a full-scale critical experiment and by extensive hydraulic and heat-transfer tests of typical reactor components and mockups. This would reduce the testing between core installation and full power to confirmation of results obtained from tests made elsewhere. An additional and very strong reason for pursuing such a course was that the reactor had been designed without much room for correcting errors found during startup. (For example, if the fuel loading had turned out to be inadequate, it would not have been simply a matter of installing more fuel elements, as in some reactors, because no additional space was left. Instead, it would have been necessary to remove some of the fuel elements, then to build and insert more heavily loaded elements in their place—a costly and time-consuming process.)

Once the broad outlines of the test program had been determined on the basis of the above considerations, the detailed planning began.

The program as planned consisted of six groups of tests, of which five groups were identified as Periods I through V, and one group as Periodic

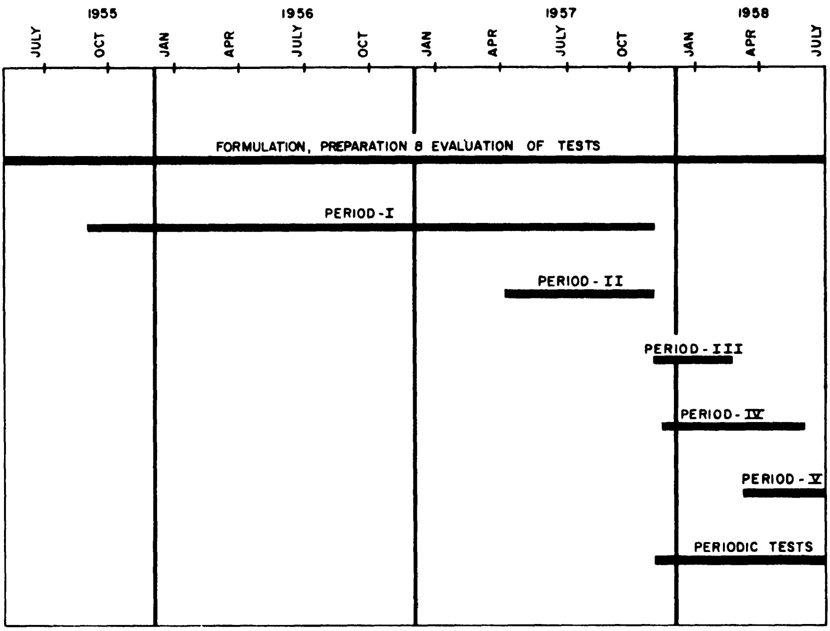

FIG. 18-1. Test periods.

Tests. Periods I through III were to prove that the plant could be started up safely. Period IV was intended to show that the plant met design objectives. Period V is intended to gather additional information on plant design and performance and to ascertain the margins, if any, by which the design requirements were surpassed. The Periodic Tests were prescribed to assure that certain plant systems and equipment would be rechecked regularly. Although the periods indicated the approximate order in which the tests were to be run, there was some overlapping. For example, Period II tests on some systems were being run at the same time as Period I tests on others. The scope of each test period, including test specifications, is given below. Figure 18-1 shows the chronological arrangement of the test periods.

18-2.2 The test periods. *Period I: Material and equipment-erection and installation.* This period covered inspecting and testing the installation of plant mechanical and electrical equipment. This included receiving inspection and cleaning material and equipment, radiographing and inspecting welds, hydrostatic testing of installed piping and pressure vessels, and calibrations of nonnuclear instrumentation. Also inspected for proper installation were electrical system wiring, cables, and components; fluid system piping and components in both the reactor and turbine-

generator plants; and ventilation and air-conditioning ducts. Each reactor plant fluid system was flushed with reactor coolant grade water to ensure system cleanliness and a free flow path. Much of this testing—for example, checks to ensure that motors rotated in the proper direction—was in accordance with normal construction practice, without detailed written instructions.

Specifications used for tests included in this period follow.

Test specifications

1. Packaging, shipping, and receiving of PWR control rod drive mechanisms.
2. General specifications for shipping, receiving, inspection, storage, and maintenance of PWR power plant equipment.
3. Unpacking, field shop testing, and storage of instrumentation.
4. General specifications for installation, checking, and inspection of PWR power plant electrical equipment.
5. Index of references for welding and weld cutting apparatus.
6. General specifications for installation, checking, and testing of PWR power plant mechanical equipment.
7. Fluid systems line flushing.
8. Fluid system and components hydrostatic testing.

Period II: Precritical systems testing. This period consisted of operational tests of individual systems and of the reactor plant without the core before initial critical operation. A series of over-all plant conditions of temperatures and pressures was determined. The test of each system was planned in several sections, each section requiring only a single plant condition. It was thus possible to perform as a group parts of several tests which required identical plant conditions. Tests during this period served principally to locate and correct deficiencies. They were valuable also for operator training and as a checkout of plant operating procedures. In the list of the tests following, all except those marked with an asterisk were performed at least once before initial criticality.

Test specifications

1. Neutron shield tank.
2. Coolant charging system.
3. Coolant discharge and vent system.
4. Chemical addition system.
5. Coolant sampling system.
6. Component cooling water system.
7. Valve operating system.

8. Reactor coolant system.
9. Coolant pressurizing and pressure relief system.
10. Decay heat removal system.
11. Coolant purification system.
12. Safety injection system.
13. Reactor plant leak tests.
14. Reactor plant instrumentation and control system.
15. Reactor rod control system.
16. Reactor power and temperature control system.
17. Reactor protection system.
18. Nuclear instrumentation system.
19. Operational radiation monitoring system.
20. Reactor plant remote viewing system.
21. Safety radiation monitoring system.
22. Reactor plant container drainage system.
23. Radioactive waste disposal system.
24. Reactor plant container air cooling system.
*25. Core-handling equipment.
*26. Canal water system.
*27. Core removal cooling system.
*28. Radioactive material viewing system.
*29. Chemical checkout of sampling train.
30. Controlled steam relief system.
31. Charging and discharging of demineralizer resins.
32. Precritical cleanup of reactor plant systems.
33. Main hydraulic valve thermal shock.
*34. Capped valve vent test.
*35. Refueling seal to reactor vessel.
*36. Functioning of the blanket disassembly equipment.
37. Reactor plant piping movement during warmup.
*38. Reactor plant container leakage test.
39. Control air system.
40. Service air system.
41. Boiler feedwater system.
42. Condensate system.
43. Main and auxiliary steam system.
44. Condenser.
45. Fire-protection system.
46. Turbine.

Period III: Initial critical testing. This period provided the information required from the reactor core and associated control and instrumentation systems during initial critical operations before high power could be

achieved. Power for testing was restricted to that which could be dissipated without synchronizing the turbine generator with the Duquesne Light power system. Included in the scope of this period were operational tests of the reactor rod-control mechanisms, initial calibrations of nuclear instrumentation, initial fill of the live reactor with water, approach to criticality, determination of reactivity coefficients, and operation at low power. These tests provided the data necessary for initial evaluation of core performance. Following is a list of the tests specified for the period. All except those marked with an asterisk were performed at least once before the generator was synchronized.

Test specifications

1. Initial fill of live reactor with water.
2. Control rod drive mechanisms, precritical and initial critical tests.
3. Initial approach to criticality.
4. Measurement of shutdown reactivity.
5. Intercalibration of nuclear instrumentation.
6. Flow distribution across the core.
7. Flow coast-down test.
8. Core instrumentation calibration.
9. Startup and manual operation at low power.

* 10. Failed element detection and location (FEDAL) system (hydraulic).

* 11. FEDAL system (electrical).

Period IV: Power operation. This period covered testing the integrated operation of the reactor plant and the turbine-generator plant systems. The tests were intended to verify that the plant met the requirements of its design for both steady-state and transient operation. In addition, the tests provided performance data required for future operational and maintenance reference as well as practical training for the Station operating personnel. This is the first period during which the station operated as a completed central electrical generating station capable of generating electrical energy to the maximum of its design rating. The tests specified were as follows:

Test specifications

1. Station startup tests.
2. Turbine startup.
3. Governor performance tests.
4. Main unit heat-rate tests.
5. Initial radiation survey.
6. Test for shield defects.
7. Reactor coolant loop outage test.

8. Steady-state load tests (including design full-load performance).
9. Decay heat removal system.
10. Load swing tests (manual-automatic load control).
11. Reactor automatic control.
12. Loss-of-load tests.
13. Shutdown tests (including reactor plant cooldown).
14. Waste disposal system performance tests.
15. FEDAL system (operational).
16. Determination of base level of reactor coolant fission products.
17. Determination of effective neutron flux for tritium production and buildup in the reactor coolant.
18. Valve operating system drift test.

Period V: Station capability and limitation investigation. Based on the results of the first four testing periods, additional tests are being specified to determine maximum plant capabilities and limitations. There will also be special tests to determine new operating methods or to obtain data useful in design changes, future equipment development, and advanced pressurized water reactor techniques.

Listed below are examples of specifications for this phase of testing.

Test specifications

1. Station output with four-loop operation.
2. Maximum station capability.
3. Variation of plant parameters (temperature, pressure, etc.).
4. Plant unbalance.
5. Natural circulation, startup and operation.
6. Heat balance at maximum capability with half-speed main coolant pump operation.
7. Maximum rate of station startup and shutdown.
8. Measurement of reactor plant heat losses and container temperatures.
9. Radiation levels in canal area during "hot" fuel transfer and storage.
10. Steady-state hydrogen distribution.
11. Modified purification system performance.
12. Constant steam pressure control.
13. Optimum reactor power and temperature control system settings.
14. Controlled steam relief system performance.
15. Power transients resulting from introduction of cold water into core.
16. Determination of reactivity lifetime.

Periodic tests. This group provides a formal mechanism for obtaining data on aspects of plant safety and performance which are desired period-

ically through the life of the plant. The tests will also provide a means of determining possible system and component faults and malfunctioning. Below is a typical list of such periodic tests.

Test specifications

1. Reactor plant container air locks leakage and performance test.
2. Radiation levels in vicinity of purification demineralizers.
3. Nondestructive testing of stainless to carbon steel welded pipe joints.
4. Periodic reactor plant radiation survey.
5. Valve operating system performance.
6. Comparison of FEDAL monitor readings with radiochemical sample data.
7. Periodic characterization of radioactive waste disposal system effluent.
8. Metallurgical and chemical inspection for irradiated subassemblies.
9. Examination of reactor vessel internals.
10. Periodic tabulation of plant parameters.
11. Reduced temperature startups.
12. Core unbalance test.
13. Stuck-rod simulation.
14. Calibration and intercomparison of control rods.
15. Xenon transient tests.
16. Determination of pressure, temperature, and flow coefficients of reactivity.

18–3. Role of the Participating Organizations

18–3.1 Westinghouse Electric Corporation. Under its prime contract with the Atomic Energy Commission, Westinghouse had responsibility for design, construction, maintenance, operation, testing, and physical security of the nuclear portion of the plant until the time the Station was officially synchronized with the Duquesne Light Company's power utility system. At the outset of the program, Westinghouse assigned to the PWR project personnel from its plant engineering department, with supporting scientific personnel from the following categories:

Plant analysis and plant layout engineers.
Hydrodynamic and fluid systems engineers.
Instrumentation and control system engineers.
Reactor design engineers.
Reactor plant and shielding physicists.
Plant chemists.

Mechanical and electrical component designers.
Materials and process engineers.
Experimental and test engineers.

When system designs were well advanced, the design and engineering groups began planning the tests for their particular portion of the reactor plant. At the peak of preparing test specifications and reviewing preliminary test procedures, it is estimated that the equivalent effort of ten full-time engineers and scientists was required for these tasks.

In February 1957, Westinghouse established a six-man group at Shippingport (separate from the construction activity), responsible for the firm's participation in the checkout and test phase of putting the plant on the line. At the peak of testing, just before power operation, there were fifty-five in the group, although many were personnel on loan for training.

An important part of the Westinghouse group was a nucleus of personnel from the S1W Project at the Naval Reactor Testing Facility near Idaho Falls, Idaho. Their previous experience in operating that plant enabled them to contribute significantly to planning and carrying out the test program.

18-3.2 Duquesne Light Company. The Duquesne Light Company had two major roles in plant checkout and testing. As designer and builder of the turbine generator portion of the Station, Duquesne formulated the tests required to prepare that part of the installation for integrated operation with the reactor plant. In addition, they furnished test engineers and operating personnel who both wrote the detailed test procedures and performed all formal tests on the reactor plant and the turbine generator plant. After the generator was first synchronized, the Duquesne Light Company assumed responsibility for performance of the entire test program.

At the end of 1955, Duquesne assigned two engineers to train for the test program. The number gradually increased to eleven by June 1956, reached a peak of twenty-three engineers and five technicians by October 1957, then leveled off to twenty-one engineers and six technicians in early 1958. During this time two to four engineers were in training at the Naval Reactor Testing Facility.

In addition, the Duquesne engineers participated in the preparation of the test specifications, described later in this chapter. Ten engineers loaned to Duquesne from Westinghouse plant design departments assisted in preparing and carrying out tests during the period of heaviest work load.

18-3.3 Atomic Energy Commission. The Atomic Energy Commission directed the formulation and execution of the test program. In the design

and construction of the nuclear portion of the plant, it had a similar role. The AEC assumed approval responsibility for all test specifications and test procedures. The main criteria in reviewing the program were safety of equipment and personnel and operation of the plant in accordance with approved procedures and accepted safeguard practices.

The Commission contributed from its experience gained on other government reactor plants by assigning personnel with experience in design or operation of reactor plants to its resident staff at Shippingport.

18–4. Execution of the Program

18–4.1 Joint Test Group. In 1956 a Joint Test Group including representatives of Duquesne Light, Westinghouse, and the Atomic Energy Commission was established to take responsibility for the over-all direction and planning of the test program. It was comprised of the following members:

Manager, Westinghouse PWR Test and Operations Group
Chief Engineer, Duquesne Light Company, Shippingport Site
Results Coordinator, Duquesne Light Company, Shippingport Station
Site Technical Representative, Atomic Energy Commission

The group reviewed and approved test specifications and procedures, determined what results were required, established testing sequences and priorities, and approved the adequacy of the results.

Recommendations of the Joint Test Group, as well as test procedures, received review and approval by the Naval Reactors Branch of the AEC.

18–4.2 Sequence and scheduling of tests. The Joint Test Group established the priority and sequence of tests subsequent to Period I. Priority was, in general, based upon a logical sequence of startup of systems as determined by system requirements and the expected plant completion schedule. Test and operations personnel of Westinghouse and Duquesne scheduled the individual tests.

18–4.3 Execution of tests. The Period I tests were performed by construction contractor personnel, under the technical direction of Westinghouse engineers in the reactor and waste disposal plants, and under the direction of Duquesne engineers in the turbine-generator plant. There was one exception, the hydrostatic test of the entire reactor plant. Because it required operating a great deal of plant equipment simultaneously, this test was performed by Duquesne operating personnel. Construction personnel repaired any deficiencies found.

All other tests were performed by Duquesne operating personnel under the technical direction of Duquesne test engineers.

In April 1957, about seven months before initial criticality, detailed lists were made of the necessary plant piping, components, instrumentation, and electrical cabling which had to be completed for each Period II test. Priority lists were given to the construction contractors who installed the systems. The priorities could not be followed absolutely because construction interfered, but they were followed closely enough for systems to be ready for testing in approximately the desired sequence. Two engineers drew up these lists in approximately six weeks.

To follow up on the priority lists, a "Test Support Group" was formed, representing the construction contractor, Westinghouse, Duquesne, and the Atomic Energy Commission. Meeting daily, the group reviewed the status of the systems needed for test within about two weeks, and took action within their own organizations to solve problems which might prevent completion of the systems. Each day the group issued a bulletin showing required action. This bulletin was widely distributed for action throughout the participating organizations.

The inspection contractor (also the design contractor) was required to certify before each hydrostatic test that all piping had been installed according to plans and specifications, and that all weld inspection records had been reconciled and approved. Although there were some last-minute delays when inspection showed deficiencies such as interference between pipes, considerable effort in retesting was saved. This approach resulted generally in sound systems needing very little alteration or repair.

In summary, this was the planned scheme of operation: first, a system or part of a system was finished in accordance with a detailed priority list; second, it was certified as to its inspection; third, a hydrostatic test of the system was performed; and fourth, where possible this test was followed by an operational test.

Later (June 1957) an additional step was interposed. Inspection showed that when the reactor coolant piping was assembled it was not clean enough to protect small clearance valves and the core. Dirt, filings, and welding slag were found in many pipes, particularly the smaller ones. To overcome this, each auxiliary system that was to be connected to the reactor coolant system was systematically flushed with demineralized water before its hydrostatic test. Each connecting system was broken and water flushed through at above its design flow rate. Flushing was continued until the water issuing from the system showed no more particle removal upon passage through a strainer (cloth for small piping, mesh for large).

Flushing proved to be an effective way of cleaning pipes for a pressurized water plant. No difficulties with any components have yet occurred as a result of dirt. It was also obvious that flushing was essential with carbon steel piping. While only small quantities of filings, slag, and dirt were

found in stainless steel piping, much larger quantities, of the order of a pound, were found in the carbon steel piping of the safety injection system. Much of this consisted of spherical pieces of slag, a few as large as a quarter inch in diameter.

While the piping systems were being assembled, inspected, flushed, and hydrostatically tested, a parallel effort began on reactor plant instrumentation. This was done by instrumentation checkout teams. Each team usually consisted of one or two engineers, two instrument steamfitters, and an electrician. Operating three at a time, these teams checked out all the instrumentation in the reactor plant, calibrating and adjusting it. In most cases, instruments such as differential pressure cells and pressure transmitters were checked as soon as they were mounted and wired, and before they were connected by piping. As a result, there was very little difficulty with reactor plant instrumentation during startup of the plant.

After preliminary checks and hydrostatic tests of auxiliary piping were complete, a large filter having hydraulic resistance equivalent to that of the core was installed in the reactor vessel. A plain hemispherical head was installed in place of the final reactor vessel head. In this condition, the nuclear plant was filled with water and its systems tested for proper operation. The plant temperature can be raised by operating the main coolant pumps at fast speed; the power applied to the pumps is converted into heat by the resistance of the system (since no energy is withdrawn from the system). This is sufficient to raise the temperature about 50°F per hour near room temperature and about 15°F per hour near operating temperature.

After the plant was filled with water, its temperature was raised to slightly above 120°F and it was hydrostatically tested to 3750 psig. (The temperature of 120°F was chosen to prevent brittle fracture of any of the carbon steel components—the reactor vessel, the Babcock and Wilcox U-type steam generators, and the pressurizer.) Although most of the reactor coolant auxiliary systems had already been individually tested, a few were retested with the reactor coolant loops and reactor pressure vessel. System pressure for the test was provided by the hydrostatic test pump in the nuclear plant.

During all the first reactor plant system hydrostatic tests, about fifteen leaks were found—pinholes in the socket welds, including both carbon and stainless steel. No leaks were found in any butt welds.

The next phase of testing required plant conditions of 120 to 200°F and 2000 psig. To facilitate pressure control in the nuclear plant, a 2000-psig nitrogen bubble was placed in the pressurizer.

During the period of operation with the 2000 psig nitrogen bubble in the pressurizer, the thermal insulation of the reactor coolant systems was

completed, and the following operational tests were run:

Charging and discharging of demineralizer resins.
Coolant purification system.
Reactor plant control and instrumentation.
Coolant charging system.
Valve operating system.
Reactor coolant system.

Those portions of the radioactive waste disposal system needed for noncritical but high-temperature operation of the plant were pressed to completion and tested. These were the spray recycle system and a portion of the surge and decay tank system. The vent gas recycle system required for hydrogen disposal during normal plant operation was not completed; and as a temporary expedient the flash tank and blowoff tank, which were designed to receive water and hydrogen discharged from the reactor coolant systems, were pressurized with nitrogen and were provided with a vent connection to atmosphere external to the plant container.

Chemically, the reactor coolant water was kept at a pH of about 7 during this cold operation, and hydrazine was employed to scavenge oxygen. A neutral ion exchange resin (HOH) was used in the purification system demineralizer to remove soluble and insoluble matter from the reactor plant water. In the final phase of the cold period operation the neutral resin was discharged, and the purification demineralizer was charged with a resin containing lithium hydroxide in order to maintain a pH of about 10. Then hydrogen was added to the reactor coolant water and the plant was heated to operating temperature with the pumps. During plant warmup, measurements were made of pipe positions to assure that changes due to thermal expansion did not cause interference and that motions were as predicted. For about the next seven days the plant was kept at operating temperature, and the following major tests were performed:

Valve operating system.
Component cooling water system.
Reactor plant leak test.
Coolant purification system.
Plant container air cooling system.
Main hydraulic valve thermal shock.
Main and auxiliary steam system.
Pressurizing and pressure relief system.
Reactor plant instrumentation and control.
Precritical cleanup of reactor plant systems.

At the end of the seven days of hot operation, all planned testing was complete, the nuclear plant's integrity had been proven, and the analysis

of the reactor coolant chemistry indicated that the system was clean. The plant was next cooled down by venting steam to atmosphere from the main steam header. During cooldown, hydrogen was vented intermittently from the pressurizer to the blowoff tank, and reactor coolant was circulated through the pressurizer at the maximum possible rate, about 40 gpm, to permit the hydrogen in the remainder of the system to come out of solution in the pressurizer and to be vented.

When the hydrogen had been reduced to about 8 cc per liter (plant at about 240°F), the pressurizer was entirely filled with water and the plant was brought to room temperature by feeding cool water to the secondary side of the steam generators.

The loops were then drained, the plain hemispherical head and filter were removed from the reactor pressure vessel, and the plant was ready for core installation.

Inspection of the reactor vessel filter and reactor vessel showed no dirt in the reactor plant systems. All that could be found was a very thin film of magnetite (Fe_3O_4), which is normally produced in high-temperature water loops.

The period from October 1, 1957 to December 1, 1957 was devoted principally to assembly of the complete reactor. However, the test program continued. The main efforts went into testing the radioactive waste disposal system, the systems involved in reactor control, and the turbine plant and its control systems.

The reactor control system checkouts included not only the formal tests, but several weeks during which the equipment was merely kept energized to permit any failures due to heat or defective parts to show up. This precaution appears to have been worthwhile, since few such failures occurred after reactor operation began.

The radioactive waste disposal system tests required the entire time devoted to assembling the reactor. Most of this time, however, was devoted to correcting difficulties in instrumentation which showed up during the tests, preventing their completion. Unlike the nuclear plant, the waste disposal system had not been subjected to the rigorous instrumentation checkout that the nuclear plant and its auxiliaries had received. This was reflected in the difference in the speed of testing the two plants. Delays and difficulties encountered in the waste disposal plant, compared with the nuclear plant, could be attributed for the most part to the difference in handling of instrumentation checkout.

The turbine plant was physically completed on October 1, 1957, and all testing required for its operation on steam was completed on December 17, 1957.

The tests completed during reactor assembly (October 1, 1957 to December 1, 1957) were:

Condensate system.
Component cooling water.
Operational radiation monitoring.
Reactor rod control system.
Initial fill of live reactor with water.
Main and auxiliary steam system.
Boiler feedwater system.
Control air system.
Radioactive waste disposal system.

At this point, the plant was ready for the first approach to criticality. This was made with the plant at about 110°F and 450 psig. To speed data processing, supplemental nuclear instrumentation consisting of fission counters and digital count rate printers were used.

The initial approach to criticality was "rehearsed" by sending operating personnel to Bettis Plant to practice on a critical assembly there, utilizing the control rod configurations to be used at Shippingport. The reactor at Shippingport was taken critical for the first time on December 2, 1957.

The following week was devoted to a series of physics tests with the plant cold, to determine individual control rod worths as well as various bank worths, shutdown margin, and excess reactivity of the core. The final preoperational tests of the rod control system and reactor protection system were also made.

Upon the completion of cold physics testing, the plant temperature was increased slowly by using the energy from the main coolant pumps. At 250, 350, 450, and 523°F, rod worth measurements were taken to obtain the temperature coefficient of reactivity and the pressure coefficient of reactivity. An attempt was made to determine a flow coefficient, but none was detected. The heatup was also used to calibrate the thermocouples in the core instrumentation. This series of tests occupied almost an entire week. After a few last-minute repairs were made, the plant was ready for reactor power, being at operating temperature and pressure after the hot critical tests.

The purpose of the initial power test, conducted at a very low level (3 percent reactor power) with steam dumped to atmosphere, was to demonstrate the reactor plant operating characteristics in the low power range.

This test was for only a few hours, and steam was then fed to the turbine to heat it and to set it rotating at low speed. The next phase was a typical steam turbine startup, in which speed was increased slowly to check the governor control and to locate the critical speed. The turbine was brought to synchronous speed, and presynchronization electrical checks were made on the generator. On December 18, 1957 the plant was synchronized with the Duquesne system and began producing electric power.

Plant power was increased in increments of about 10 megawatts electrical output. Steady-state conditions were achieved at each increment, and heat balances, nuclear instrumentation calibrations, and station radiation surveys made. Reactivity coefficients were also obtained at each power level. Full power—60 megawatts net electrical output with three loops in operation—was reached on December 23, 1957, and the plant operated at that power level for 100 continuous hours. The plant was then shut down to test the decay heat removal system and to follow the xenon buildup and decay.

After initial power operation, testing was continued in two general, fundamentally independent, categories: those tests requiring power production such as the response of the plant to load changes, first with the plant in manual control and then in automatic control and those tests independent of plant operation, such as those on the auxiliary systems and the waste disposal plant.

The following tests were performed between completion of the initial full-power test and March 31, 1958:

Load swing tests in manual and automatic control.
Control rod positions for criticality.
Determination of coefficients of reactivity.
Control rod drive mechanisms.
Reactor plant control and instrumentation.
Radioactive waste disposal system.
Capped valve vent test.
Operational radiation monitoring system.
Reactor plant container air cooling system.
Canal water system.

18.5. Principal Lessons Learned

The experience gained in testing and putting the Shippingport plant on the line would require several volumes to record. This chapter concludes with a brief discussion of some of the lessons learned with the Shippingport plant which might be valuable to those formulating and executing test programs for other plants.

18–5.1 Plant instrumentation. The time saved and difficulties avoided by thorough checkout of instrumentation prior to plant startup cannot be overstressed. Much testing of instrumentation was done at the vendor's plant, followed by the checkout and testing after installation described previously. It is believed that a very extensive program of testing starting with rigorous performance tests by the vendor should prove to be an economical investment.

18–5.2 Piping. It is believed that the benefits of thorough flushing of reactor plant piping systems have been demonstrated by experience at Shippingport. Also, it was shown that by careful control, piping systems could be hydrostatically tested almost on an individual line basis and thus made available for systems testing at an earlier time.

18–5.3 Information. It was found useful to have detailed information in the form of component instruction books available to field forces no later than the time of delivery of the equipment. Such information gives personnel responsible for installation and testing a chance to become familiar with the equipment and to follow proper procedures in putting it into service. This tends to reduce equipment failures and malfunctioning due to improper startup. It should also lessen the strong dependence on equipment specialists and result in more competent operating and maintenance personnel.

18–5.4 Other testing and experimental work. It was found desirable to make maximum use of tests and experiments elsewhere than at the plant itself, such as critical tests, loop tests, model studies, and the like, so that pre-startup testing could be of the nature of confirmatory or safety checks rather than experiments. Insistence on predicted test results from the test designers will tend to assure sufficient preliminary work to help accomplish this end. Simulator (analogue computer) studies were also very helpful in this regard.

18–5.5 Formulation of tests. Considerable emphasis was placed on the over-all test planning. It was found helpful to subdivide all tests into sections according to the plant conditions required, and to have the test sections with each set of conditions carefully reviewed for optimum sequencing. The plant conditions established for each set of tests were carefully selected and, once established, made virtually inflexible. All persons connected with test planning were kept informed of the plant conditions that were established.

Finally, it was concluded that there is no unique set of tests or sequence which must be observed for a satisfactory startup. There are always many ways to achieve a particular goal. It was found to be important to establish a sequence of tests and hold to it as closely as possible. When it was absolutely essential to make changes, considerable effort was made to insure that all concerned were informed. Communication, always a management problem, is more than ever important in a fast-moving test program.

Supplementary Reading

1. PWR—The Significance of Shippingport, *Nucleonics* **16**(4), 53–72 (1958).
2. Shippingport Issue, *Westinghouse Engr.* **18**(2) (March 1958).
3. A. L. Bethel et al., *Shippingport Atomic Power Station Inspection and Test Program*, USAEC Report WAPD-PWR-972, Westinghouse Atomic Power Division, 1957.
4. W. H. Hirst, *PWR Inspection and Test Program*, USAEC Report WAPD-PA-210, Westinghouse Atomic Power Division, Dec. 14, 1956.

CHAPTER 19

PROJECT CONTROL AND PROCUREMENT

CHAPTER 19

PROJECT CONTROL AND PROCUREMENT*

19–1. INTRODUCTION

Project control and material procurement for the Shippingport Atomic Power Station offered challenges to the personnel involved, in much the same way that the design and development work presented a challenge to the scientists and engineers of the project. Not only were there the usual problems involved in control and procurement for a major construction program conducted on an accelerated schedule, but also the radically different types of equipment developed to meet new and exacting requirements necessitated creating a new industry capable of supplying such equipment.

The PWR control and procurement effort was unusual in two principal ways. First, nearly all components of the plant were supplied by contractors, rather than being manufactured at the Westinghouse Bettis Plant. This situation posed problems in both scheduling and quality control. Second, as already mentioned, the requirements for most components were out of the ordinary in one or more ways. Often the materials called for were unfamiliar to work with, and specifications were rigid. Many components were physically much larger than any that had been heretofore constructed of these materials. Hydraulic components had to be absolutely leaktight, and many moving parts had to be able to operate for very long periods without maintenance. Tolerances were often such that new techniques had to be developed to meet them, and it is believed that this fact has significantly aided in advancing not only the nuclear power program, but also other industrial developments.

The control and procurement functions in support of design and construction were handled at the Westinghouse Bettis Plant principally by four groups: Planning and Control, Purchasing, Manufacturing Engineering, and Quality Control. Needs for equipment originating with the design engineers were presented to Planning and Control, who coordinated scheduling for the whole project; this group continued to coordinate the efforts of the various Bettis groups until procurement was completed, providing constant follow-up on schedules throughout the duration of plant design and construction. Planning and Control transmitted equip-

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* By J. C. Rengel and M. A. Penfield, Westinghouse Bettis Plant, and R. V. Laney, U. S. Atomic Energy Commission.

ment needs to the Purchasing Department, who had responsibility for carrying through the purchasing process, including schedule follow-up with vendors. Purchasing procured the equipment, materials, and services required within the cost and time limitations imposed. Manufacturing Engineering advised design engineers on product designs and specifications, evaluated supplier capability to produce the items to specification in the required length of time, and assisted suppliers in manufacturing problems. Quality Control assured that components were produced by the vendors according to specification quality requirements.

This chapter will discuss these functions.

19–2. Coordination

The problems of coordination involved in design, development, and procurement were divided into two principal categories: (1) scheduling and the necessary follow-up to ensure that the schedules were met, and (2) ensuring that the procured items met requirements. The first function was the responsibility primarily of Planning and Control for internal control and of Purchasing for supplier control. The second function was the responsibility of Government inspection agencies, of the Quality Control organization, and of the design engineers.

19–3. Planning and Scheduling

The responsibility of Bettis Planning and Control began with the establishment of target dates. For example, in July 1953, a target date of March 1957 was established for completion of construction. In addition, full-power operations were to be achieved by the end of 1957. All subsequent planning and scheduling were based on these targets, with schedules being developed backwards from them. Planning and Control had to coordinate the scheduling of engineering with that of procurement. This coordination was begun by a systems planning group organized for this purpose. Forms necessary for scheduling control were developed, and 42 "control points" for each major component, forming the basic schedule structure, were set up. The number of control points was later reduced to 26. The basic master form, listing the control points, is shown in Fig. 19–1. Before detailed scheduling of components was possible, however, completion of system design descriptions had to be scheduled (Fig. 19–2).

A basic ground rule agreed to between the Westinghouse Bettis Plant and the AEC was that no procurement on any items of a system could commence until the system description was approved by the AEC.

In scheduling engineering and procurement, attention was directed first to systems (such as the reactor coolant system) in which the dif-

OFFICIAL USE ONLY	SYSTEM —	F R.—
S O —	COMPONENT —	E-SPEC.-
P.O —	VENDOR —	QUANTITY-
BUYER —	RESPONSIBLE ENGR'S.	

EVENT	REQ'D DATE	COMP. DATE	DRAWING INFORMATION
F. R. TO COMP. ENGR.			
E-SPEC. ISSUED			
E-SPEC. TO PAO - APPROVAL			
E-SPEC. APPROVED			
E-SPEC. TO MI TO PURCHASING			
INQUIRY TO PROSPECTIVE VENDORS			
PROPOSALS DUE			
QUOTES TO MI FOR RELEASE OF P.O.			
PAO APPROVAL OF ORDER			
PLACE PURCHASE ORDER			
PLAN LIST & OUTLINE DWG.			
PLAN LIST & OUTLINE DWG. TO PAO			
PAO COMMENTS & DWG. APPROVAL			
ASSEMBLY & DETAIL DWGS.			
ASSEMBLY & DETAIL DWGS. TO PAO			
PAO COMMENTS & DWG. APPROVAL			
WELDING PROCEDURES APPROVED - WAPD			
FABRICATION PRODEDURES APPROVED			
MFG. RELEASE			
PRODUCTION SCHEDULE			
CLEANING SPECS. APPROVED			
COMPLETION OF MFG.			
TEST AT CLAIRTON SITE			
SHIP TO SITE			
INSTRUCTION BOOKS			
SPARE PARTS			

REMARKS

VENDOR PROM DELIVERY	WAPD ESTIMATE DELIVERY	
WHEN SEPARATED FROM ENCLOSURES HANDLE THIS DOCUMENT AS OFFICIAL USE ONLY	WESTINGHOUSE ELECTRIC CORPORATION BETTIS ATOMIC POWER DIVISION PGH, PA. PROJECT FOR PRIMARY PLANT MASTER SCHEDULE	SCHEDULED BY ISSUE DATE SHEET NO

FIG. 19-1. Primary plant master schedule.

SECTION I

PWR SYSTEMS REQUIRING PA/NRB APPROVAL

PWR PRP-11-Issue No

S O	System	First E Spec Approved	Last E-Spec Approved	System Description Submitted to NRB	Date System Approved by NRB	Final Submission Date to NRB
1 AP-8004	REACTOR PLANT CONTROL					
1-AP-8036	FAILED ELEMENT DETECTION & LOCATION					
1 AP-8057	CORE INSTRUMENTATION					

* DENOTES COMPLETION DATE

SHEET 1

SECTION II

PWR REACTOR DESIGN SYSTEMS

PWR-PRP-11-Issue No

S O	System	Design Description Issue Date	System Description To PAO	First Drawing Approved By NRB	Last Drawing Approved By NRB
1-AP-8001	HEAT SOURCE SYSTEM				
1 AP-8002	REACTIVITY CONTROL SYSTEM (MECHANISMS)				
1-AP-8003	REFUELING AND HANDLING EQUIPMENT				

# DENOTES COMPLETION DATE

SHEET 2

SECTION III
PWR CONSTRUCTION DESIGN (S & W)
PWR PRP-.1 Issue No

S O	System	Design Criteria Issue Date	Preliminary Drawings From S & W	WAPD Review And Comment	Final Construction Drawings From S & W	Final Construction Drawing To PAO For Approval
8020	REACTOR PLANT CONTAINER DRAINAGE					
8032	INTERNAL SHIELDING—AUXILIARY CHAMBER BOILER CHAMBER					
8035	REACTOR PLANT CONTAINER					
8046	NEUTRON SHIELD TANK & REACTOR					
8047	REACTOR & FUEL CANAL CONCRETE					
	UPPER STRUCTURE OVER REACTORS					
8050	REACTOR FUEL HANDLING SERVICE BUILDINGS AND AUXILIARY EQUIPMENT STRUCTURE					

* DENOTES COMPLETION DATE

SHEET 3

PWR APPROVED SYSTEMS
PWR PRP-11-ISSUE NO

S O	SYSTEM	FIRST E-SPEC APPROVED	LAST E-SPEC APPROVED	FIRST SYSTEM DESCRIPTION TO NRB	DATE SYSTEM APPROVED BY NRB	DATE OF FINAL SUBMISSION TO NRB
8005	MAIN COOLANT					
8006	REACTOR PRESSURIZING & PRESSURE RELIEF					
8007	COOLANT PURIFICATION					
8008	COOLANT CHARGING					
8009	COOLANT DISCHARGE & VENT					
8011	DECAY HEAT REMOVAL					
8012	COOLANT SAMPLING					
8014	CHEMICAL ADDITION					
8015	COOLANT VALVE OPERATING					
8016	SERVICE AIR					
8017	COMPONENT COOLING					
8019	REACTOR PLANT CONTAINER AIR COOLING					
8020	REACTOR PLANT CONTAINER DRAINAGE					
8021	REACTOR ROD CONTROL					
8022	REACTOR POWER AND TEMPERATURE CONTROL					
8022	CONTROL STEAM RELIEF					
8023	NUCLEAR INSTRUMENT					

* DENOTES COMPLETION DATE

SHEET 4

PWR APPROVED SYSTEMS
PWR-PRP-11-ISSUE NO

S O	SYSTEM	FIRST E-SPEC. APPROVED	LAST E-SPEC. APPROVED	FIRST SYSTEM DESCRIPTION TO NRB	DATE SYSTEM APPROVED BY NRB	DATE OF FINAL SUBMISSION TO NRB
8024	OPERATIONAL RADIATION MONITORING					
8025	REACTOR PROTECTION					
8025	REACTOR PROTECTION (ALTERNATE)					
8026	PRIMARY PLANT CONTROL					
8027	PRIMARY PLANT INSTRUMENTATION					
8028	PRIMARY ELECTRICAL POWER					
8029	SAFETY RADIATION MONITORING					
8030	REMOTE VIEWING					
8033	REACTOR PLANT MISCELLANEOUS UTILITIES					
8040	STEAM PIPING					
8041	BOILER FEED					
8043	WASTE DISPOSAL					
8045	CORE REMOVAL COOLING					
8048	THERMAL INSULATION					
8053	SAFETY INJECTION					
8055	CONTROL AIR					
8056	CANAL WATER					

* DENOTES COMPLETION DATE

SHEET 5

FIG. 19-2. Systems schedule and design status report (5 sheets).

S O NO	SYSTEM COMPONENT	E-SPEC	P O NO	QTY	VENDOR	COST

ENGINEER				STATUS
DELIVERY DATE DATA				
PROM - W	W TEST	PROM - SITE	REQD - SITE	
CONTRACT DEL				
W ESTIMATE				

ENGINEER				STATUS
DELIVERY DATE DATA				
PROM - W	W TEST	PROM - SITE	REQD. - SITE	
CONTRACT DEL				
W ESTIMATE				

ENGINEER				STATUS
DELIVERY DATE DATA				
PROM. - W	W TEST	PROM. - SITE	REQD.- SITE	
CONTRACT DEL				
W ESTIMATE				

ENGINEER				STATUS
DELIVERY DATE DATA				
PROM. - W	W TEST	PROM - SITE	REQD - SITE	
CONTRACT DEL				
W ESTIMATE				

SHEET 1

ITEMS RECEIVED AT SHIPPINGPORT POWER PLANT

SYSTEM OR S O NO	COMPONENT	P O NO	QTY	RECEIVED

SHEET 2

FIG. 19-3. Component status report (sheets 1 and 2 of 2).

STATUS OF PWR REPAIR PARTS PROCUREMENT

	NUMBER	PER CENT
TOTAL NUMBER OF ORDERS RECEIVED AT SITE * TOTAL NUMBER OF ORDERS REQUIRING PROCUREMENT ACTION TOTAL NUMBER OF ORDERS REQUIRING SPARE PARTS		
* BREAKDOWN OF ORDERS REQUIRING PROCUREMENT ACTION		
NUMBER OF ORDERS IN MANUFACTURE NUMBER OF ORDERS IN PURCHASING FOR PLACEMENT NUMBER OF ORDERS AT AEC FOR APPROVAL NUMBER OF RECOMMENDATIONS BEING REVIEWED BY ENGINEER NUMBER OF RECOMMENDATIONS NOT RECEIVED FROM SUPPLIER ** TOTAL UNCOMPLETED ORDERS		
** BREAKDOWN OF DELIVERY PROMISES OF TOTAL UNCOMPLETED ORDERS		
PERIOD 1 2 3 INDEFINITE ORDERS UNPLACED TOTAL UNCOMPLETED ORDERS		
NUMBER OF ORDERS PARTIALLY RECEIVED-__________ THESE ORDERS ARE NOT INCLUDED IN TOTAL OF COMPLETED ORDERS		

SHEET 1

REPAIR PARTS STATUS — VENDOR RECOMMENDATIONS
SYSTEM

S O #	COMPONENT	VENDOR	ORIGINAL P O #	E-SPEC

SHEET 2

REPAIR PARTS STATUS — DELIVERY PROMISES
SYSTEM

S O #	COMPONENT	VENDOR	ORIG PO #	E-SPEC	R P PO #	PO PLACED	PROM TO SITE

SHEET 3

REPAIR PARTS STATUS — ITEMS RECEIVED
SYSTEM

S O #	COMPONENT	VENDOR	ORIG PO #	E-SPEC	RO PO #	PLACED	REC'D AT SITE

SHEET 4

FIG. 19-4. Repair parts status report (sheets 1, 2, 3, and 4 of 4).

ficulty of designing and fabricating components caused the longest "lead times" (times between initiation of procurement and delivery of component). Among the factors that made fabrication of reactor coolant system components difficult were the large size of the components, their special requirements, and the new techniques being used.

Detailed scheduling and internal follow-up on components were done by Planning and Control in cooperation with Project Design Engineering, Purchasing, Manufacturing Engineering, and the AEC. With Project Design Engineering, a date was worked out for each component on which the equipment specification would be completed; with the AEC, a date for AEC approval of the specification; and with Purchasing and Manufacturing Engineering, the lead time for the component. In this way, the detailed scheduling involving the 26 control points was prepared. Planning and Control assisted Purchasing in obtaining a minimum scheduled lead time by supplying them with preliminary information that permitted them to begin negotiations with suppliers as soon as possible.

Early in the planning and scheduling process, a list of particularly critical items, as determined by experience, was submitted to Manufacturing Engineering, whose help was to be needed on such items. This list included the reactor vessel, the steam generators, the main coolant pumps, the reactor coolant system valves, and much of the instrumentation. For many such items, manufacturing engineers were assigned to the suppliers' plants as resident engineers. Their firsthand knowledge applicable to the problems encountered by the suppliers assisted greatly in expediting delivery dates.

The documentation involved in the scheduling program was basically in three parts: (1) the master schedules, (2) status reports on components, and (3) status reports on instruction manuals and spare parts. Two of the basic forms used are shown in Figs. 19-3 and 19-4. It was particularly important to have instruction manuals in ample time because they greatly aided in installing components, preparing operating instructions, and training plant operators. Availability of spare parts on schedule was important because items damaged or misplaced during construction could cause serious delays if replacements were not available. It was also important to have spares available for ready replacement during the plant testing program.

19-4. Buying

At the beginning of the PWR program in 1953, the PWR project was dependent upon a centralized plant Purchasing Department also serving other AEC and Navy projects for which Westinghouse was responsible.

Later, the decision was made to divorce from this centralized purchasing department the responsibility for procurement for nuclear cores. The Bettis Nuclear Core Department assumed this responsibility for the PWR core. By January 1957, the Purchasing Department had been reorganized, and the procurement of PWR project items other than nuclear cores became the responsibility of a new PWR Purchasing Department.

Buyers were faced with procuring such items (many of them as yet undeveloped) as a 250-ton reactor vessel, an 80-ton pressurizer, control rod drive mechanisms, fuel extraction tools, core hoisting equipment, nuclear instrumentation, hermetically sealed pumps, leaktight valves, and large diameter, seamless, heavy walled, stainless steel piping. All these items were not only larger than normal, but were of specialized design (described in detail in other chapters). Finding suitable suppliers was consequently a difficult problem. Ordinary avenues of procurement were often inapplicable for the following reasons:

(1) The components included items never previously manufactured.

(2) Even the items of standard manufacture included modifications, features, and/or requirements that were new.

(3) Many potential suppliers were doubtful of the advisability of undertaking the risks involved in this new field.

(4) Rigid specifications and lack of firm designs for some items at the time of soliciting quotations did not present an attractive picture to potential suppliers.

(5) Peculiar problems existed in connection with establishing suitable facilities and accountability control for certain special nuclear materials.

To alleviate these difficulties, extensive efforts were made to create an understanding of the need for rigid specifications for the component design requirements and the future market for such components. Personal visits were made to suppliers' plants by appropriate Bettis engineers and procurement personnel to evaluate facilities and create an interest in participation in the program. In addition, all Bettis Plant suppliers were sent a brochure describing the Bettis activities and organization, atomic power, and the future market for atomic components, and they were invited to visit Bettis facilities for orientation. Manufacturers were encouraged to discuss Bettis requirements as well as to examine drawings, models, and samples of the type of products required.

It was also difficult to find suppliers qualified to develop and produce specific items. Investigations were made in related fields of endeavor and in unrelated industries whose manufacturing processes and products bore a similarity to PWR requirements. The value of this investigation is illustrated by the fact that control rod drive mechanisms were furnished by organizations such as an aircraft landing gear manufacturer and an

automotive transmission producer, and that the extraction tool was developed and built by a producer of shoe machinery.

One of the specific problems encountered was to locate a sufficient number of valve suppliers who were in a position to produce valves meeting the particular requirements of the plant. This difficulty was chiefly due to the fact that most valve producers were equipped to manufacture valves primarily on a mass production basis. Valve procurement was further complicated by a lack of testing facilities to test some of the valves at PWR design temperature and pressure.

One interesting problem handled by Purchasing was negotiation for and establishment of a relief valve test facility. This facility was established with the excellent cooperation of the American Gas and Electric Company.

Relief valves employed in the reactor coolant systems of a pressurized water reactor plant are fundamentally important components. Because of the requirements of very high corrosion resistance, variable back pressure for reseating, and nearly zero leakage, special designs had to be employed. Also, such valves must operate under extreme thermal stress conditions, at high differential pressures and flow rates. For these reasons, it was imperative to prove designs for Shippingport relief valves by testing prototypes under the most stringent conditions that might be encountered in operation. In addition, all production units had to be tested to ensure satisfactory operation after installation, especially since they were to be welded into the systems.

Since both the steam and water relief valves were much larger than any such valves previously produced, no suitable test facilities existed. A source that could supply steam and water at sufficient flow rates, temperatures, and pressures had to be found, and a test loop for valve testing constructed to utilize it. In 1955 a Bettis Plant survey team began a search for electric utility power plants that could fulfill the requirements. A survey of eleven power companies yielded three plants that could produce adequate test conditions. Of these, two were the property of companies unwilling to undertake such a venture. The third was the Twin Branch plant of the Indiana and Michigan Electric Company, a subsidiary of the American Gas and Electric Company; this plant is located at Mishawaka, Indiana. The utility company agreed to the test loop and it was constructed under the conditions that (1) plant operating requirements would take priority over valve testing needs, (2) the company would have final approval in regard to designs, equipment, and personnel connected with the test loop, and (3) the company could terminate the agreement on one year's notice. Design and construction required 12 months.

The test loop is capable of supplying to valves to be tested 800 gpm of 650°F water at 3000 psig or 80,000 lb/hr of saturated steam at 2300 psig.

All relief valves employed in the PWR reactor coolant system were tested in this loop.

Whenever possible, equipment procurement was done on the basis of firm price purchase actions, while research and development work was contracted for by means of cost type subcontracts, usually on a cost plus fixed fee basis.

It sometimes became necessary to undertake work before a firm design could be established. Further, the design would be influenced by the manufacturing facilities and techniques of the manufacturer. As a result, a subcontract was drawn on a cost plus fixed fee basis for the development and design of the item, and a firm price basis for manufacture. The price for manufacture was established when the item reached a stage that an accurate cost estimate could be made. This type of subcontract was used on the reactor vessel and closure head.

Even under normal buying conditions the amount of time required to issue a purchase order is an important factor. For PWR, this situation was aggravated by very tight schedules. The requirement of Government contracting officer approval of procurement actions above established monetary levels increased the time necessary to execute large purchase orders and subcontracts.

19-5. Supplier Follow

Detailed knowledge of the status of each order placed was necessary to facilitate a realistic evaluation of delivery dates and to enable construction field forces to schedule the number and types of personnel and equipment required at Shippingport.

It was necessary, in some instances, to rearrange the sequence of construction to better conform with delivery of components. A form (Fig. 19-5) indicated, for example, the date a supplier would submit outline drawings, welding and cleaning procedures, detailed drawings, repair parts recommendations, and instruction books. Shortly after an order was placed, a Purchasing Department coordinator checked with the supplier to establish realistic submission dates for such documents. This procedure permitted establishment of a complete manufacturing release date and an evaluation of sufficient lead time for the supplier to meet the construction schedule.

During manufacture, evaluations of progress were made by Purchasing and the supplier. For example, typical steps performed at a foundry to produce a stainless steel body casting which would meet the requirements for a large valve are enumerated below. Although the coordinator did not require dates for all the intermediate steps, it was necessary that he follow the processes in order to evaluate the delivery date.

1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31

SUPPLIER		PROJECT	PO NO
E-SPEC	DESCRIPTION		PO DATE
QUANITY	SHIPPING SCHEDULE		SO NO

MONTH										
SCHEDULE										
COMPLETE										

MONTH										
SCHEDULE										
COMPLETE										

NO	EVENT	SCHEDULE		PROMISE					COMPLETE C/N	COMPLETE DATE
1	PLAN LIST									
2	OUTLINE/ASSY. DRAWINGS									
3	OUTLINE/ASSY. DWGS. APPROVED									
4	PARTIAL MANUFACTURING RELEASE									
5	REPAIR PARTS FORM TO VENDOR									
6	DETAIL DRAWINGS									
7	DETAIL DRAWINGS APPROVED									
8	REPAIR PARTS LIST FROM VENDOR									
9	WELDING PROCEDURES									
10	WELDING PROCEDURES APPROVED									
11	WELDING SAMPLES									
12	WELDING SAMPLES APPROVED									
13	WELDABILITY SAMPLES									
14	WELDABILITY SAMPLES APPROVED									
15	CLEANING, ASSY. & FAB. PROCEDURES									
16	CLEAN., ASSY. & FAB. PROCEDURES APPVD.									
17										
18										
19										
20										
21	COMPLETE MANUFACTURING RELEASE									
22	INSTRUCTION SHEETS/BOOKS (PRELIM.)									
23	INSTRUCTION SHEETS/BOOKS (APPVD.)									
24	INSTRUCTION SHEETS/BOOKS (FINAL)									
25	COMPONENT WEIGHED WEIGHT									
26	FINAL DRAWINGS									
27	COMPONENT SHIPMENT									

SHEET 1

EVENT NO 6 – DETAIL DRAWINGS									
DRAWING NO	TITLE	SCHEDULE	PROMISE					REC'D	APPROVED

SHEET 2

FIG. 19–5. Procurement form (sheets 1 and 2 of 2).

(1) Pattern.
(2) Mold.
(3) Pouring.
(4) Cleaning.
(5) Finish grinding.
(6) Preparation of casting for dye penetrant examination.
(7) Dimensional check.
(8) 100% dye penetrant examination.
(9) Radiography.
(10) Removal of defective areas.
(11) Re-radiography of areas to determine that all defects have been removed.
(12) Repairing of defects by welding.
(13) Re-radiography of repair-welded areas.
(14) Heat treatment.
(15) Final dimensional check.
(16) 100% x-ray (final x-ray required by Westinghouse for approval).
(17) 100% dye penetrant inspection.
(18) Buyer's inspection.
(19) Shipment.

19–5.1 Planning and control meetings. In July 1955 the PWR project initiated a policy whereby weekly control meetings would be held to review all major problem areas or critical items that were delaying the progress of the program.

The agenda for the first meeting, held on July 27, 1955, included the following items:
(1) Reactor vessel manufacturing.
(2) Space and capital equipment requirements.
(3) PWR inspection policies and procedures.
(4) Procurement and testing of relief valves.
(5) Plan for off-site radiation monitoring program.

The meetings were attended by the Bettis Project Manager, design engineers, Planning and Control, Purchasing, and appropriate AEC personnel. These weekly control meetings reviewed the existing problems and established a plan of action for resolving the problems. Once the plan of action was established, individuals were assigned responsibility to carry it out. Starting with the meeting of September 26, 1955, the PWR Planning and Control Group assumed the responsibility for preparing the agenda. The procedure used for pinpointing major problem areas for discussion at the weekly meetings was as follows.

Planning and Control reviewed the progress of the design and procurement phases against the established schedules and noted those cases which

were running behind schedule and which would affect the end completion date of the plant.

At the end of 1955, it became apparent that the core and plant could be separated for project control purposes. Therefore, the control groups were segregated on January 1, 1956. Core Planning and Control was made responsible for monitoring and preparing the weekly agenda covering the reactor design, core manufacturing, and associated programs; Plant Planning and Control was made responsible for monitoring and preparing the agenda covering the design and procurement of the power plant systems and components.

Each of the Planning and Control Groups reviewed the progress and status of work on the system designs, initiation of equipment specifications, and the procurement or manufacturing status of components, indicating the major problems which required special action, and included these latter items in the weekly agendas. Group review and action assignment meetings were held weekly, starting in January 1956 and continuing through November 1957.

To ensure that no item would be overlooked and later turn out to be a bottleneck, monthly meetings were held to review the status of every item appearing on the PWR Plant Bill of Material. Plant Planning and Control originated a work sheet for each item appearing in the Bill of Material and noted the latest status of all the required actions for each component on the individual work sheets. These work sheets were provided the meeting attendees in advance.

These monthly meetings were attended by the Bettis Project Manager and representatives from Engineering, Purchasing, Plant Planning and Control, and the AEC. They extended over an eight-month period (April–December 1956), at which time the number of items became small enough that they could be reviewed in the weekly meetings.

19–5.2 Quality control. Quality control on components for the plant was the responsibility of (1) the supplier who fabricated the component, (2) Government inspection personnel, (3) the Westinghouse Bettis Plant Quality Control organization, and (4) the design engineer, especially where extensive testing was required or a waiver of specifications was being considered.

The function performed by Westinghouse Bettis Plant Quality Control is best described by the general definition of the organization's responsibility: "To assure Bettis management and the Government that the completed item has met all specification quality requirements as set forth in the supplier contract." This responsibility is carried out by "field inspection," in the supplier's plant, by Bettis Quality Control personnel. The PWR project utilized the services of Quality Control for all "special"

components for the plant. Quality Control normally did not perform field inspections on standard commercial items. Quality Control reviewed every purchase requisition and decided whether the item would be subject to field inspection.

At the time that Quality Control was called upon to consider a particular item, the equipment specifications for this item were submitted to them for their review and comment. One of the most troublesome problems in the quality control process occurs if there is a lack of adequate understanding by the supplier of the requirements for the item to be produced. An important factor in avoiding this problem is precise specifications. For this reason, Quality Control reviewed all equipment specifications carefully and recommended changes where it was believed that a misunderstanding might occur. Similarly, drawings were also reviewed by Quality Control and suggestions made regarding any changes deemed advisable. Liaison was maintained as necessary between Quality Control and the design engineers for each item on the PWR Project. Another important quality control function consisted of interpreting and explaining to the supplier, at the initiation of the contract, the quality requirements on the item to be produced and the reasons for these requirements.

Suppliers were notified at the time bids were solicited whether Westinghouse field inspection would be performed at their plant. Some contractors considered such inspection likely to impose requirements beyond those they considered necessary for the job, while others welcomed it as assistance that facilitated their effort.

Field inspection ensured that proper procedures were being followed in each step of the manufacturing process, thus preventing wasted time and effort that might result from deferring inspection until the item was completed. It also provided assurance that proper inspections and tests would be performed on the completed item before it reached Shippingport, where such inspections and tests would in many cases have been impracticable and repairs or adjustment expensive or time-consuming.

During the most active phases of the PWR procurement program resident engineers or inspectors were established in other cities to facilitate quality control. With their families, men moved to new locations to perform field inspection duties at one or more suppliers' plants in the various areas.

19-6. Planning and Scheduling of Construction

The efforts of the contractors to coordinate and plan the construction work in the nuclear portion of the plant left many voids in the channels of communications that affected the proper scheduling and expediting efforts of Westinghouse and subcontractors not under the direct control of Dravo Corporation.

Biweekly progress meetings were therefore initiated on April 3, 1956, attended by Westinghouse site personnel and key contractors' personnel. The Westinghouse Site Manager functioned as chairman. As problems became more numerous and required timely evaluation and reports on their status, these meetings were rescheduled to once a week starting August 6, 1956. The agenda was distributed prior to the meeting to afford those attending an opportunity to come fully prepared and thus hold the discussion period to a minimum. The agenda typically followed this outline:

(1) Construction status—brief report by the construction superintendent.

(2) Critical items and major problems—joint review and planned action by Westinghouse and/or Dravo.

(3) Plans for the coming week—discussion of construction schedule, manpower, working hours, etc.

Westinghouse Production and Engineering representatives participated in the discussions as required.

To coordinate the efforts of all construction contractors at the site, the Field Coordinating Committee, consisting of the field managers and superintendents for Westinghouse, Dravo, Duquesne Light Company, and Burns and Roe, met at regular intervals. An AEC representative was the chairman of this group. Seemingly impossible scheduling situations in a small area on an exceedingly tight schedule were resolved through intensive planning and with full cooperation of all contractors involved.

With construction approximately 90% completed (six months before the end of construction) the emphasis was changed from area construction to the prosecution of work in accordance with system check lists so that those items required for earlier tests were completed and checked out first. This early planning did much to assist the field supervisors in laying out their work more effectively and in understanding the importance of concentrating work earlier in construction than normal procedure would indicate. Subsequent events proved that this action was beneficial to the timely completion of the construction and testing programs.

With construction about 95% complete, a Test Support Group was activated, consisting of line supervisors from Westinghouse, Dravo, Duquesne Light Company, and Stone & Webster. This group issued daily bulletins noting system deficiencies. Deficiencies listed by the Test Support Group were given highest priority by the construction forces. Items unresolved at the time of the regular weekly progress meetings were identified and corrective action relative to planning, production, scheduling, or necessary overtime was initiated.

With construction about 97% complete, brief daily meetings between key construction personnel and key test and operations personnel were

held to integrate construction and test work. Without these meetings, the goal long looked forward to—"Full Power in 1957"—would not have been achieved.

19–7. Conclusion

The importance of the planning, scheduling, and follow-up program cannot be overemphasized. The scheduling program provided, at each stage of the design, procurement, and construction phases of the project, comprehensive data on the status of the work. Thus any problems that threatened to delay progress could be pinpointed and a clear picture presented of what had to be done to circumvent them. In addition, it is believed that the thorough, over-all "systems" approach utilized in the initial planning contributed significantly to producing a better plant in a shorter time.

The experience gained in the procurement program for the Shippingport Atomic Power Station has emphasized the importance of developing an industry capable of supplying components for nuclear reactor plants. Future nuclear power plant programs will be able to profit from this experience as well as from the experience of the PWR project's suppliers, as these programs carry nuclear plant technology forward to more advanced states of development.

The successful completion within the scheduled period of the large procurement effort required for the Shippingport nuclear plant is a consequence of the cooperative efforts of the Government-Industry team.

Supplementary Reading

1. H. Mason, *Selection and Application of Materials for the PWR Reactor Plant,* USAEC Report WAPD-PWR-971, Westinghouse Atomic Power Division, 1957.

2. Atomic Load, *Pennsy.* **5**(10), 4–6 (November 1956).

CHAPTER 20

ORGANIZATION OF THE STATION AND PREPARATION FOR OPERATION

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To operate any power plant, the first problems that must be faced are to prepare an organization, select personnel, and establish a training program. At Shippingport, with no precedents to follow and with few nuclear-trained personnel within the Duquesne Light Company organization, the task was especially difficult. Further complications were added by the over-all program objectives—the station was to serve both as a test facility and as an operating unit of the Duquesne Light Company system. All these factors had to be considered in planning the personnel program.

20-1. Selection of Personnel

In obtaining personnel, the Company adhered to past policy—that persons already employed within the Company be considered before outside sources were utilized. Except for some recent college graduates and certain specialists, such as the Health Physicist, all personnel were obtained from within the organization. A main factor considered when a person was selected was that he must have a flexible outlook so that he would readily accept new methods and techniques.

20-2. Training of Personnel

The next step was to determine the general training requirements for those assigned to the station and to find the facilities at which this training could be obtained. In these matters, the Atomic Energy Commission and the Westinghouse Bettis Atomic Power Division rendered valuable assistance with knowledge gained in their experiences with similar problems. Training at AEC installations included formal classroom training, on-the-job training at the station, and inspection trips to equipment test and assembly facilities.

Most of the training was carried out at the Atomic Energy Commission's Naval Reactor Facility at Idaho Falls, Idaho, which is operated by the Westinghouse Bettis Atomic Power Division. This choice was made principally because a training program was already in operation which agreed closely with the training requirements established for Shippingport.

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Forty-eight station employees received a total of 171 man-months of training in periods varying from 2 to 11 months. On-the-job experience was given the men in reactor plant chemistry, health physics, maintenance, operation, instrumentation and control, and testing.

A few employees also received training in health physics and chemistry during periods of several weeks at the Materials Testing Reactor. Nuclear instrument and control engineers received approximately six months of formal and on-the-job training at the AEC's Savannah River Project.

Formal and on-the-job training was also given at the Bettis Plant. Seven members of the station organization received training in various phases of work during one-year working assignments with the PWR project. Assignments varied, but included work in the reactor design, the construction and operation of the PWR critical experiment, the operation and maintenance of the Bettis Test Facility, and design work in shielding, fluid systems, and plant analysis. Formal training in radiochemistry, a two-month course covering theory and laboratory practice, was conducted by Bettis personnel for six station chemists. The Health Physicist spent six months in the Health Physics Group at Bettis, directly involved in the Shippingport off-site monitoring program. Several shift reactor engineers received practical experience in working assignments at the Bettis Plant during the initial assembly of the PWR core.

Further formal classroom training for the men consisted of four courses of a general nature, not specifically related to the Shippingport station. The courses were chosen to give station personnel a working background in particular subjects with which they were not entirely familiar either by experience or previous education. The courses, for the most part conducted by faculty from local institutions, were presented in a manner similar to conventional college courses. Classes in nuclear physics, servomechanisms, and reactor technology were given technical personnel during the first six months of 1956 and were repeated in 1957, except that an applied electronics course was substituted for servomechanisms. Also, the amount and degree of instruction given by station supervisors to other personnel in on-the-job training prompted a course of instruction on teaching methods.

On-the-job training in practically all phases of operation and in some phases of maintenance, including the operation of maintenance and fuel-handling equipment, was conducted at the station by supervisory personnel. This phase of training began during the spring of 1957, continued through station startup, and is continuing during station operation. It consists of an orientation program for new employees and a series of lecture courses and demonstrations. At the start it also included the actual operating experience obtained during preliminary operation.

The station orientation program consists of about four lecture hours and a tour of the site which is given to each employee as he reports to work,

or shortly thereafter. It is an introduction to the station, covering a short history of the PWR, the Duquesne Light Company—AEC agreement, a breakdown of contributing organizations, and descriptions of the station cycle, the station layout and facilities, the station organization, employees' job duties and responsibilities, and station procedures.

The lecture courses cover a wide field of subjects, most of which pertain specifically to the Shippingport station. Some exceptions are elementary theory courses given to nontechnical personnel.

The specific lectures given to each employee depend on the scope and level of work in which he is involved. Lectures on security and health physics are given to all employees at regular intervals. In conjunction with these lectures, operating personnel (operators and attendants) are required to complete training evaluation records or qualification check lists on both reactor plant and turbine-generator plant systems. Operators (reactor, turbine, and switchboard) are trained on all systems; attendants are required to become proficient only on those systems covered by their job descriptions.

Lectures and demonstrations (except for those of a general nature, such as station orientation, security, and health physics) are usually given by the employee's immediate supervisor. Operating training is given by the shift reactor engineer and the shift foreman; since only one or two operators or attendants on a shift can be spared for training at any one time, this phase of instruction is usually on a teacher-pupil basis. To standardize the scope and degree of material covered, lecture outlines or question and answer sheets are prepared and distributed to the supervisors.

Pre-startup training in reactor and reactor plant operation was provided by means of a simulated reactor control console in abbreviated form. It is equipped with conventional hand controllers from which station operating setups can be made. An electronic computer receives signals from the console and returns simulated operating data and other indications to the console instrumentation, alarms, and indicating lights, which duplicate actual station operating conditions. The simulator design was based on station parameters and system component functions. Numerous operations can be duplicated, including emergency conditions and transients that can be remotely introduced into the equipment to test the operator's ability to cope with actual operating conditions.

Much operating training was acquired in preparing Volume I of the station *Operating Guide*, whose 51 chapters cover both system and station operating procedures. Procedures were written by Duquesne Light Company operating supervisors and reviewed by Westinghouse and AEC engineers.

Permanent maintenance personnel were on the job long enough before the core was installed to receive training in assembling the core and head

and in operating fuel-handling and reactor maintenance equipment. These employees, as well as the shift reactor engineers and other maintenance supervisory personnel, were integrated into the Westinghouse core assembly crew for training. Several shift reactor engineers were also integrated into the organization responsible for preliminary assembly of all head-mounted equipment (cranes, extraction equipment, alignment jigs, etc.) during test operations on those items.

All welders employed in constructing the station, including those to be permanently employed at Shippingport, received instructions in shielded inert gas nonconsumable electrode welding of stainless steel and aluminum, and were expected to meet Duquesne Light Company welding qualifications. Station welders also received welding experience on mockups of major welds.

Inspection trips were integrated with formal classroom training and on-the-job training whenever conditions made them worth while. Included were observations of the following processes:

Manufacture, assembly, and testing of reactor instrumentation and control equipment.

Manufacture of fuel elements and fuel element subassemblies.

Manufacture, assembly, and testing of control rod mechanisms.

Assembly of reactor core and core instrumentation.

Testing of fuel and core-handling equipment.

Testing of reactor assembly and maintenance equipment.

20-3. Organization

The following is a breakdown of the organization of the Shippingport Atomic Power Station, and details of the training given each individual.

The organization (Fig. 20-1) consists of five functional groups, each headed by a supervisor reporting to the Station Superintendent.

The station is headed by the *Station Superintendent,* who is responsible for the operation and maintenance of the station and the safety of all

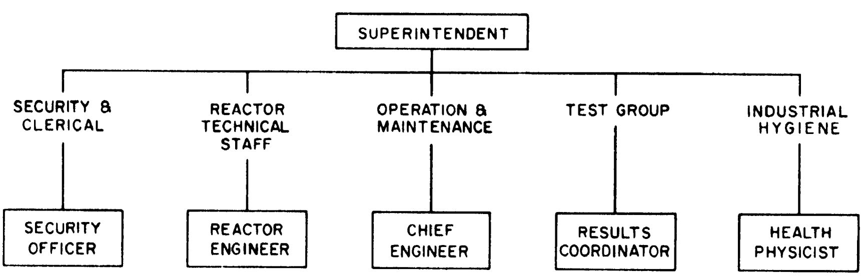

Fig. 20-1. Supervisory staff.

personnel and equipment. He reports directly to the Company's General Superintendent of Power Stations.

The *operating and maintenance group,* headed by the Chief Engineer, is responsible for operation and maintenance of the entire station. This group is somewhat similar to the corresponding staff of a conventional power station.

The *test group,* headed by the Results Coordinator, is responsible for all test work and, as will be shown later, for chemical control of the station. They perform all tests, prepare all test reports, and assist in evaluating new methods of operation and control. This group has its counterpart, on a reduced scale, in a conventional station.

The *reactor technical staff,* headed by the Reactor Engineer, is responsible for technical assistance to the operating, maintenance, and test development groups insofar as the reactor plant is concerned, and is responsible for the safety of the reactor plant.

The *industrial hygiene group,* headed by the Health Physicist, is responsible for radiation safety. Its members are required to perform routine radiation surveys of personnel, station work areas, and adjacent off-site areas. They maintain comprehensive records of these surveys.

The *security and clerical group,* headed by the Security Officer, is responsible for the over-all security of the station, including the security of classified documents and fissionable material. They process all visitor permits and, when necessary, provide escort service for visitors. The clerical portion of this group performs the duties of a similar group in a conventional station, such as maintaining station records, cost figures, etc.; they also aid in performing security duties.

To carry out these duties, a total of 132 employees is necessary. Of this total, 37 are technically trained. While this number might appear high in comparison with similar groups in conventional stations, it is justifiable for the following reasons:

(1) The station, the first of its kind, requires that more personnel, especially those technically trained, be available to cope with the new techniques involved.

(2) More men are needed for the extensive test program and the increased emphasis on health physics and security.

(3) A pool of technical manpower, trained in operating the station, must be maintained to compensate for the normal manpower turnover.

Other items which add to the manpower requirements will be pointed out in the following articles, which describe how personnel are distributed among the various functional groups.

20-3.1 Operation organization. The *Chief Engineer* is responsible for the operation of the station; in the absence of the Superintendent, he

assumes the latter's responsibilities. He receives technical assistance in the operation of the reactor plant from the *Reactor Engineer*, who is also responsible for reactor plant instrumentation and control and reactor safety.

A *station operating engineer* is in charge of each shift. He is responsible to the Chief Engineer for the safe and efficient operation of the station and, in the absence of the station staff, is responsible for the security of the station and the safety and actions of all personnel.

The *shift foreman* directs the actual operation of the station. He ensures that all operations are conducted safely and according to the approved instructions, that operating records are kept properly, and that maintenance clearances (declaring that components or systems are out of service for maintenance or again ready for operation following maintenance) are properly made. Except during terms of test, he will regulate station output in accordance with system demands.

The *shift reactor engineer* provides technical assistance to the station operating engineer. He is responsible to the Reactor Engineer for the safe operation of the reactor and, prior to each startup, checks out the reactor controls and makes necessary control rod calculations.

The operation of the station is divided into three areas of responsibility: the reactor, turbine, and electric plants, with an *operator* assigned to each. The operators control their respective plants from a central control room. They perform all major operations, execute maintenance clearances, and maintain operating records. They are assisted in these duties by the *attendants*, who also operate locally controlled equipment, record routine data, and lubricate and clean equipment.

The qualifications and specific training of the personnel in the operating group are as follows:

(1) The *Chief Engineer* is a graduate engineer with an extensive background in power station operations. He received training for five months in the operation, maintenance, and testing of reactors and reactor plant auxiliaries.

(2) The *Reactor Engineer* is a graduate engineer with power station experience and a sound background in electronics. He received training for five months in reactor operation, control, and instrumentation.

(3) The *station operating engineers* are graduate engineers with power station experience. They received training in power station operation, if they were not already qualified in that phase, and received five months of training in reactor plant operation.

(4) The *shift foremen* are high-school graduates with extensive experience in power station operation.

(5) The *shift reactor engineers* are experienced graduate engineers. They received training for four months in reactor operation and control.

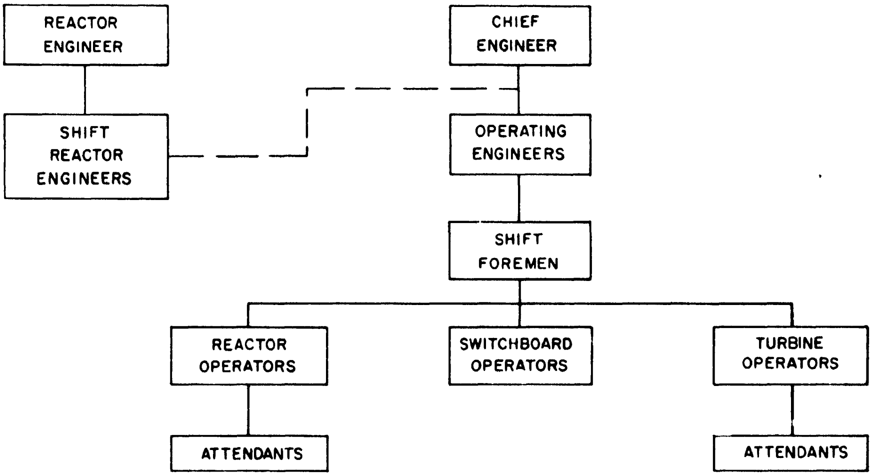

FIG. 20-2. Operation organization.

(6) The *plant operators* have a high-school education and have a sound basic understanding of power station operation. They received training for two months in actual reactor operation and on-the-job training. All were given the same training to permit the interchange of operating assignments.

(7) The *attendants* have a high-school education and are familiar with power station operations. They received on-the-job training and were included in the general training of the station.

An organization chart of the operation group is shown in Fig. 20-2.

The *Chief Engineer* is also responsible for station maintenance, receiving technical assistance from the Reactor Engineer on matters pertaining to the reactor.

The *Maintenance Engineer* is responsible to the Chief Engineer for all site and station maintenance. He directs the work of the equipment foreman and the instrument and control engineers in scheduling and planning routine maintenance and refueling operations. During core changes he is assisted by the shift reactor engineers.

The *equipment foreman* directs all maintenance, including site service activities, such as janitorial services.

Normal maintenance activities are performed by the maintenance repair group, which includes mechanics, electricians, and welders. Instrument maintenance and calibration are performed by the nuclear instrument and control engineers and technicians and the mechanical instrument repairman. The latter maintains the turbine plant mechanical instruments, while the nuclear instrument and control personnel maintain all reactor plant instruments and controls. House and yard laborers handle all site service

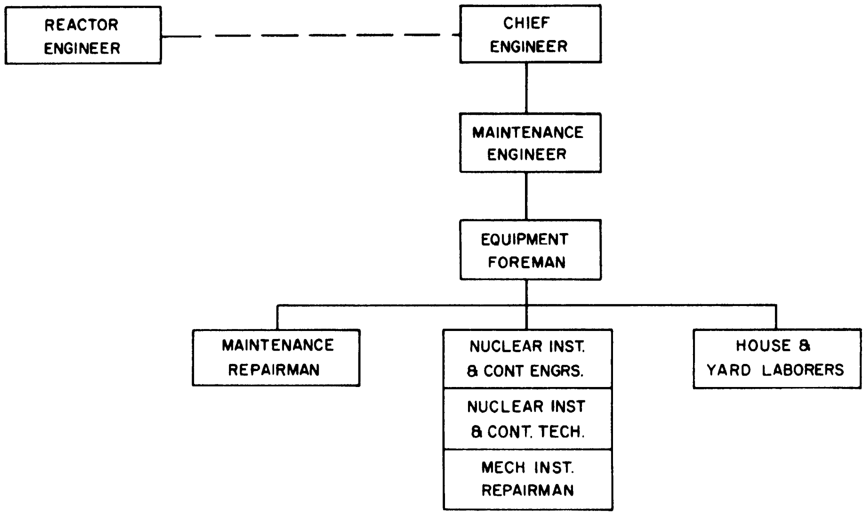

FIG. 20–3. Maintenance organization.

activities, including the operation of the reactor plant laundry. The maintenance group is supplemented during core changes and other heavy maintenance load periods by personnel from Duquesne Light Company stations.

The number of personnel in the maintenance group is higher than that required in a conventional station because the possible presence of radioactivity in some areas requires that a high degree of station cleanliness be maintained. More repairmen are required because radioactivity levels could limit the amount of time that a man can work on reactor plant equipment. The relative inaccessibility of this equipment, due to the plant containers, and the requirement of absolute leaktightness also add to the manpower requirements. The requirements would be higher still if the reactor plant equipment necessitated maintenance comparable to that of coal handling and firing equipment.

The need for instrument and control engineers and technicians was dictated by the large amount of remote control and the complicated nature of the reactor plant instrumentation and control. In addition, the extensive test program required a great deal of precise instrumentation not normally found in a power station.

The qualifications and specific training for the members of the maintenance group are as follows:

(1) The *Maintenance Engineer* is a graduate engineer with extensive experience in power station maintenance. He received five months of training in reactor plant maintenance and fuel handling.

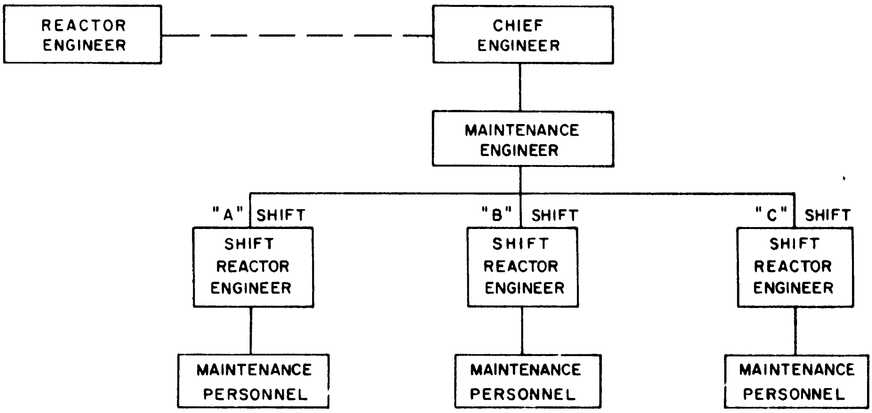

FIG. 20–4. Maintenance organization for reactor overhaul and core change.

(2) The *equipment foreman* is required to have the equivalent of a college education, and experience in power station work, stressing electrical maintenance. He received training in the maintenance of a reactor plant, including instrumentation and control.

(3) The *nuclear instrument and control engineers* are either graduate engineers with electronic backgrounds or graduates of recognized industrial electronic schools. They received approximately nine months of training in maintenance and calibration of reactor plant instrumentation and controls.

(4) *Nuclear instrument and control technicians* are high-school graduates and also graduates of a technical school specializing in electronics. These technicians received additional specialized on-the-job training.

(5) The *maintenance repairmen* and the *mechanical instrument repairman* are high-school graduates with experience in power station maintenance. With the exception of the welders, who received specialized training, they received on-the-job training and were included in the general training at the station.

(6) The *house and yard laborers* are high-school graduates. They were also included in the general training program.

Organization charts of the maintenance group are shown in Figs. 20–3 and 20-4.

20–3.2 Test organization. The *Results Coordinator* administers the station test program and chemical operations. He determines, with the assistance of the plant designers and station operating personnel, the type and scope of tests to be performed on the turbine-generator portion of the station, and he serves in a coordinating and advisory capacity in regard

to the tests on the nuclear portion of the plant as prescribed by the Atomic Energy Commission. He is also responsible for the initial evaluation of the test results. He is assisted in matters pertaining to the reactor by the Reactor Engineer.

The *Efficiency Engineer* supervises the preparation of test procedures, the performance of tests, and the preparation of reports. He schedules the actual testing in accordance with the station operating conditions.

The *test engineers* prepare the test reports and also assist in setting up special instrumentation whenever necessary.

The *test technicians* assist in the over-all program, performing calibrations, taking test data, and preparing curves.

The *Reactor Control Chemist* is responsible to the Results Coordinator for the chemistry of the entire station and advises him as to the type and scope of chemical tests to be performed. He supervises the preparation of chemical test procedures, the performance of these tests, and the preparation of test reports. He also prepares decontamination procedures and supervises major decontamination operations.

The *Radiochemist* is responsible for monitoring and collecting all radioactivity in the reactor coolant system and determining its source. He also performs the analyses required in connection with the radioactive waste disposal system. In addition, he prepares test procedures, conducts tests, and prepares test reports.

The *chemists*, under the direction of the Reactor Control Chemist, are responsible for the chemistry of the station. They assist the Health Physicist in the routine chemical analyses required for the reactor plant.

This test group is much larger than would be the case were it not for the extensive test program planned for the first few years of operation. The program is designed to check the performance of the station and to answer many questions aimed at improving the design of this and future stations.

The qualifications and training for the members of this group are as follows:

(1) The *Results Coordinator* and the *Efficiency Engineer* are graduate engineers with extensive power station test experience. Each man received training for a full year in all phases of reactor plant testing.

(2) The *test engineers* are graduate engineers with varying amounts of test experience. They received from one to six months of training in reactor plant testing.

(3) The *test technicians* are high-school graduates with little, if any, power station experience. They received on-the-job training and were also included in the general training at the station.

(4) The *Reactor Control Chemist* is a graduate chemist with experience in radiochemistry. He received two months of training in radiochemical laboratory work and two months of training in reactor plant chemistry.

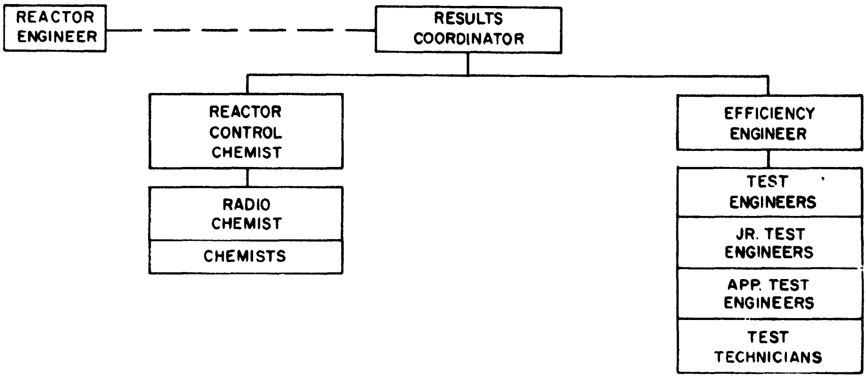

FIG. 20-5. Test organization.

(5) The *Radiochemist* is required to have essentially the same qualifications as the Reactor Control Chemist; he received the same training.

(6) The *chemists* are college graduates with little, if any, experience. They received two months of training in laboratory work and two months in reactor plant chemistry.

An organization chart of the test group is shown in Fig. 20-5.

20-3.3 Industrial hygiene group. The *Health Physicist* is responsible for the radiation safety of the station. He plans routine and special radiation surveys, and determines allowable exposure times and protective clothing requirements for personnel who work in areas where radiation exists. He ensures that adequate records are kept of radiation surveys, exposure of station personnel and visitors, and radioactive materials transferred to or from the station.

The *radiation control technicians*, working on a shift basis, perform survey work, determine personnel exposure from dosimeter and film badge readings, and ensure that radiation monitoring equipment is in satisfactory operating condition. They also assist in directing the work of the house and yard laborers assigned to the reactor plant areas, and oversee the operation of the personnel change room.

The *radiation field technicians* process samples collected by the radiation control technicians and the chemistry group, to determine count rates. They also assist the radiation control technicians in the performance of their duties.

This group, which is peculiar to reactor plants, possibly will be required in future stations of this nature, since the results of excessive radiation exposure are quite severe.

It is necessary to maintain extensive official records of personnel exposure and of radiation surveys made at regular intervals in the areas surrounding the site to protect against radiation damage claims that may

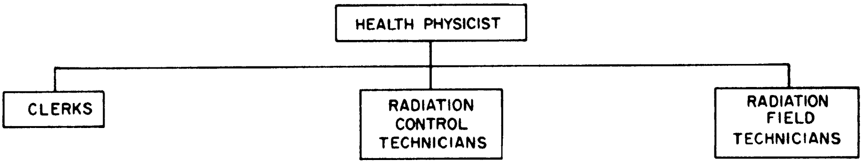

FIG. 20–6. Industrial hygiene group.

arise. In view of the latter, a program was initiated to determine the background radiation in the Shippingport area and in the waters of the Ohio River. This information serves as a reference for the surveys being made during station operation.

The qualifications and training of the personnel in this group are as follows:

The *Health Physicist* is a college graduate with knowledge of the health physics problems involved in the maintenance of power station equipment. He received extensive training in health physics at the Naval Reactor Facility.

The *remainder of the personnel* in this group are high-school graduates. They received on-the-job training in their particular jobs and were included in the general training program at the station.

An organization chart of the industrial hygiene group is shown in Fig. 20–6.

20–3.4 Security and clerical group. The *Security Officer* is responsible for the security of the station, including documents and fissionable materials, and for the direction of the station clerical group. He issues all visitor permits and initiates personnel clearances.

The physical security of the station is maintained by the *guard force.* They admit all personnel and visitors to the station and provide visitor escort service when required. Two guards are on duty at all times, and a third is provided during the day for visitor escort and administrative duties.

The *clerical group* performs duties similar to those of the clerical personnel of a conventional power station. They also provide classified document control and assist in other security duties.

This group is also much larger than would normally be the case in a conventional power station, because of (1) the tighter security requirements occasioned by the presence of fissionable materials and the large number of visitors who tour the plant, which means considerable paperwork for control of documents and visitor permits, (2) the large number of station personnel, and (3) the necesssary stenographic services connected with the test program.

The qualifications and training of the personnel in this group are as follows:

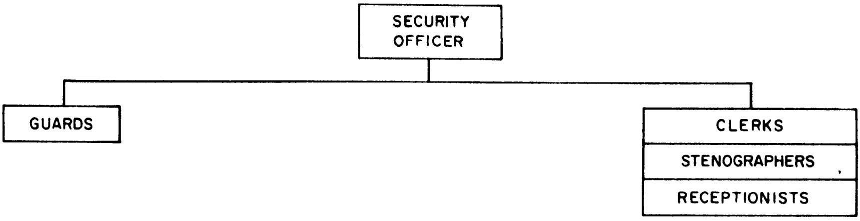

FIG. 20–7. Security and clerical group.

The *Security Officer* is a high-school graduate with a sound knowledge of power station administrative procedures. He received training in physical security methods and in classified document and fissionable material control.

The *clerical group* has qualifications similar to those for conventional station personnel and was included in the general training program at the station.

An organization chart of the security and clerical group is shown in Fig. 20–7.

20–4. Schedule of Personnel Buildup

Figure 20–8 shows the buildup of the organization at Shippingport before operation as compared with that of a conventional station of essentially

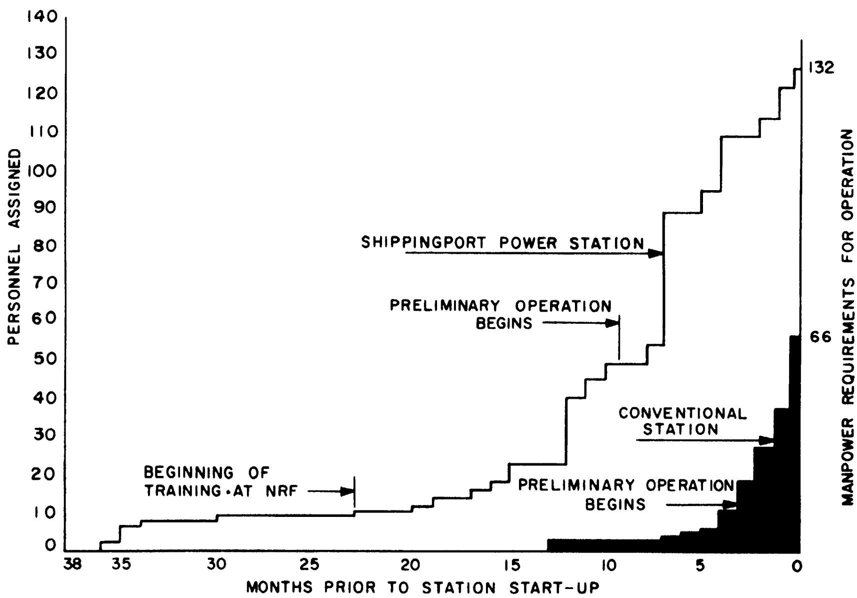

FIG. 20–8. Personnel assignment schedule; Shippingport station vs. conventional station.

equal capacity. The extensive training program, the preparation of the Operating Manual, and the preliminary work associated with the test program are the major factors contributing to the number of personnel required. However, the extended period of preliminary operation, as indicated on the chart, was also a factor.

20–5. Conclusion

On December 2, 1957, the Shippingport Atomic Power Station reactor was taken critical for the first time. Its output was synchronized on December 18, 1957, and electric power produced by atomic nuclei flowed into the Duquesne Light Company system. From that time on, the station has been operating at loads dictated by either the Duquesne Light system demand or the testing program.

The entire operation to date has been satisfactory. It is believed that the organization developed, the employee selection, and the training outlined above are responsible for this successful performance.

As mentioned before, one of the principal values of Shippingport to the atomic power program is that it provides, from practical experience, an evaluation of the design and construction features used.

Another function of the station is to determine whether the best operating and maintenance organization and training programs have been utilized. If it is determined after considered evaluation, which can only come from operating experience, that changes should be made, this will be done. It is felt, however, at least for the present, that a satisfactory program has been achieved.

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APPENDIX

SHIPPINGPORT PLANT AND REACTOR CHARACTERISTICS

(Based on a net generated electrical output of 60 Mw with three coolant loops operating)

A. General Parameters	
1. Reactor power, thermal	225 Mw
2. Gross electrical output	68 Mw
3. Net electrical output	60 Mw
4. Reactor coolant system operating pressure	2000 psia
5. Steam pressure—full load	600 psia
6. Reactor coolant average temperature	523°F
B. Reactor Coolant System	
1. Flow per loop	7.62×10^6 lb/hr
2. Loop cold-leg temperature	508°F
3. Loop hot-leg temperature	538°F
4. Main coolant pump power requirements	1250 kw/pump
5. Main coolant pump head	112 psi
6. Main coolant pump capacity	19,300 gpm/pump
7. Reactor vessel pressure drop	40.5 psi
8. Coolant loop pressure drop	72 psi
9. Total coolant volume	2500 ft^3
10. Coolant volume in core	103 ft^3
11. Steam generator heat-transfer area	
Babcock & Wilcox unit	9100 ft^2
Foster Wheeler unit	8300 ft^2
12. Steam generator over-all heat-transfer coefficient	
Babcock & Wilcox unit	827 Btu/(hr)(ft^2)(°F)
Foster Wheeler unit	992 Btu/(hr)(ft^2)(°F)
13. Steam flow, full load (total)	860,000 lb/hr
C. Auxiliary Systems	
1. Coolant purification system	
a. Flow rate (total)	40,000 lb/hr
b. Demineralizer resin operating temperature	120°F
c. Demineralizer decontamination factor (based on 15-min activity)	25:1

2. Charging system

a. Hydrostatic test pump,	head	4500 psia
	capacity	2 gpm
b. Charging pump,	head	3000 psia
	capacity	25 gpm/pump
c. Fill pump,	head	100 psia
	capacity	200 gpm
d. Storage tank capacity		50,000 gal

3. Pressurizing system

a. Pressurizer temperature	636°F
b. Allowable positive pressure surge—normal operation	2180 psia
c. Allowable negative pressure surge—normal operation	1850 psia
d. Pressurizer volume	261 ft^3
e. Pressurizer heaters	342
f. Pressurizer heater rating (per heater)	2500 watts

4. Radioactive waste disposal limits

a. Maximum activity of liquid discharged (exclusive of tritium)	10^{-8} $\mu c/ml$
b. Maximum activity of gas discharged (exclusive of tritium)	4×10^{-7} $\mu c/ml$
c. Quantity of liquid discharged (exclusive of tritium):	
maximum daily average	1590 μc/day
maximum per day	6200 μc
d. Quantity of tritium discharged:	
maximum daily average	10 curies/day
maximum per day	50 curies

D. Chemistry Specifications

1. Reactor coolant

a. Oxygen, max	0.14 ppm
b. Hydrogen	25 to 50 cm^3/kg
c. Chlorides, max	0.1 ppm
d. pH	9.5 to 10.5

2. Secondary system water

a. Phosphate (as $PO_4^{\equiv}$)	100 to 300 ppm
b. Sulfite (as $SO_3^{=}$)	25 to 100 ppm
c. Chlorides, max	0.3 ppm
d. Total dissolved solids, max	1000 ppm
e. pH	10.6 to 11.0

E. Maximum Radiation Levels (Design Limits)	
1. Reactor coolant loop compartment; loop shut down, adjacent loop operating	200 mr/hr
2. Auxiliary container (outside component shields)	50 mr/hr
3. Turbine room, control room	2 mr/hr
No continually manned area over 2 mr/hr	
F. Turbine-Generator and Steam Plant	
1. Turbine-generator design rating, gross	100 Mw
2. Turbine operating speed	1800 rpm
3. Main transformer rating	115,000 kva
4. Condenser surface area	70,000 ft^2
5. Condenser rating (based on 114,000 gpm of circulating water at 85°F)	817×10^6 Btu/hr
G. Core Assembly Characteristics	
1. Configuration	Right cylinder
2. Size	6.8 ft dia. × 6 ft high
3. Fuel load	
a. Seed	
U^{235}	75 kg
b. Blanket	
UO_2	16.0 tons
Natural uranium	14.2 tons
4. Composition	
a. Seed	
Water	52.6% volume
Fuel alloy	13.3% volume
Zircaloy	34.1% volume
b. Blanket	
Water	43.5% volume
UO_2	30.0% volume
Natural uranium	14.55% volume
Zircaloy	26.5% volume
5. Control Rods	
a. Configuration	Cruciform
b. Number	32
c. Material	Hafnium
d. Total rod worth	25.6% $\Delta k/k$
e. Absorber length	$71\frac{1}{2}$ in.
f. Weight of control rod and adapter	55 lb
g. Scram time	0.35 sec time delay 1.0 sec rod drop

6. Control rod drives	
a. Type	Collapsible rotor mechanisms
b. Number	32
c. Power requirements	220 volts, 3 phase
d. Position indication	
Accuracy, inverter reluctance type indicator	$\pm\frac{1}{8}$ in.
Accuracy, magnetic coil indicator	$\pm 1\frac{1}{2}$ in.
7. Fuel elements	
a. Seed	
Type	Plate
Number	32 assemblies
Plates per assembly	60
Channel thickness	0.069 in.
Plate thickness	0.069 in.
Meat dimensions	2.050 in. × 0.039 in. × 70.75 in.
Meat composition	
U^{235}	6.3%
Zircaloy	93.7%
Cladding	Zircaloy-2
Plate temperature	
Average	580°F
Maximum	740°F
Flow area	2.20 ft^2
Active heat-transfer surface	3855 ft^2
Weight of zirconium	4.50 tons
b. Blanket	
Type	Rod
Number	113 assemblies
Bundles/assembly	7
Rods/bundle	120
Rod dimensions	0.411 in. dia. × $10\frac{1}{4}$ in.
Fuel pellet dimensions	0.3575 in. dia. × 0.3494 in.
Fuel	UO_2 (dioxide of natural uranium)
Cladding	Zircaloy-2
Weight of zirconium	7.93 tons
Fuel temperature	
Average	1000°F
Maximum	3800°F

Flow area	8.62 ft^2
Heat-transfer surface	7912 ft^2

H. Nuclear Data

1. Seed	
a. Percent of fissions by neutrons with energy above 0.6 ev	10%
b. Percent of fissions by neutrons with energy below 0.6 ev	90%
c. Thermal neutron flux	
Average	5.7×10^{13} $n/(cm^2)(sec)$
Maximum	1.9×10^{14} $n/(cm^2)(sec)$
d. Prompt neutron lifetime	5.6×10^{-5} sec
e. Effective delayed neutron fraction	0.0077
f. Temperature coefficient of reactivity	
Hot (525°F)	-3.1×10^{-4} $\Delta k/°F$
Cold (110°F)	-0.5×10^{-4} $\Delta k/°F$
2. Blanket	
Percent of fissions by neutrons with energy above 0.6 ev	17%
Percent of fissions by neutrons with energy below 0.6 ev	83%
Thermal neutron flux	
Average	4.4×10^{13} $n/(cm^2)(sec)$
Maximum	2.5×10^{14} $n/(cm^2)(sec)$
3. Control requirements	
a. Burnup	11.0% $\Delta k/k$
b. Equilibrium xenon	3.1% $\Delta k/k$
c. Equilibrium samarium	0.7% $\Delta k/k$
d. Temperature defect	2.6% $\Delta k/k$
e. Total	17.4%
4. Heat flux values (3 loops)	
a. Seed	
Average	98,400 $Btu/(hr)(ft^2)$
Maximum	418,000 $Btu/(hr)(ft^2)$
b. Blanket	
Region 1 (center)	
Average	52,000 $Btu/(hr)(ft^2)$
Maximum	195,000 $Btu/(hr)(ft^2)$
Region 2 (inside, and adjacent to, seed)	
Average	82,900 $Btu/(hr)(ft^2)$
Maximum	343,000 $Btu/(hr)(ft^2)$

b. Blanket (*continued*)

Region 3 (outside, and adjacent to, seed)	
Average	58,200 Btu/(hr)(ft^2)
Maximum	322,000 Btu/(hr)(ft^2)
Region 4 (outermost)	
Average	30,800 Btu/(hr)(ft^2)
Maximum	141,000 Btu/(hr)(ft^2)